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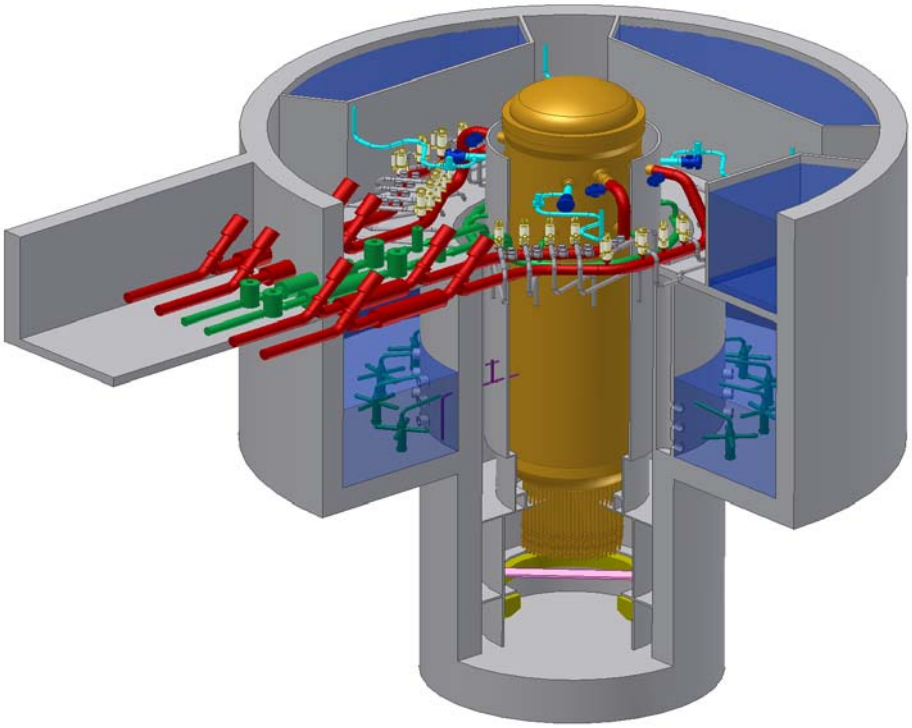

ESBWR Design Control Document

Tier 2

Chapter 5

Reactor Coolant System And Connected Systems

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Global Abbreviations And Acronyms List

Term	Definition
10 CFR	Title 10, Code of Federal Regulations
A/D	Analog-to-Digital
AASHTO	American Association of Highway and Transportation Officials
AB	Auxiliary Boiler
ABS	Auxiliary Boiler System
ABWR	Advanced Boiling Water Reactor
ac / AC	Alternating Current
AC	Air Conditioning
ACF	Automatic Control Function
ACI	American Concrete Institute
ACS	Atmospheric Control System
AD	Administration Building
ADS	Automatic Depressurization System
AEC	Atomic Energy Commission
AFIP	Automated Fixed In-Core Probe
AGMA	American Gear Manufacturer's Association
AHS	Auxiliary Heat Sink
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
AL	Analytical Limit
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AOV	Air Operated Valve
API	American Petroleum Institute
APLHGR	Average Planar Linear Heat Generation Rate
APRM	Average Power Range Monitor
APR	Automatic Power Regulator
APRS	Automatic Power Regulator System
ARI	Alternate Rod Insertion
ARMS	Area Radiation Monitoring System
ASA	American Standards Association
ASD	Adjustable Speed Drive
ASHRAE	American Society of Heating, Refrigerating, and Air Conditioning Engineers
ASME	American Society of Mechanical Engineers
AST	Alternate Source Term
ASTM	American Society of Testing Methods

Term	Definition
AT	Unit Auxiliary Transformer
ATLM	Automated Thermal Limit Monitor
ATWS	Anticipated Transients Without Scram
AV	Allowable Value
AWS	American Welding Society
AWWA	American Water Works Association
B&PV	Boiler and Pressure Vessel
BAF	Bottom of Active Fuel
BHP	Brake Horse Power
BOP	Balance of Plant
BPU	Bypass Unit
BPWS	Banked Position Withdrawal Sequence
BRE	Battery Room Exhaust
BRL	Background Radiation Level
BTP	NRC Branch Technical Position
BTU	British Thermal Unit
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CAV	Cumulative absolute velocity
C&FS	Condensate and Feedwater System
C&I	Control and Instrumentation
C/C	Cooling and Cleanup
CB	Control Building
CBHVAC	Control Building HVAC
CCI	Core-Concrete Interaction
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CIRC	Circulating Water System
CIS	Containment Inerting System
CIV	Combined Intermediate Valve
CLAVS	Clean Area Ventilation Subsystem of Reactor Building HVAC
CM	Cold Machine Shop
CMS	Containment Monitoring System
CMU	Control Room Multiplexing Unit
COL	Combined Operating License
COLR	Core Operating Limits Report
CONAVS	Controlled Area Ventilation Subsystem of Reactor Building HVAC
CPR	Critical Power Ratio
CPS	Condensate Purification System
CPU	Central Processing Unit

Term	**Definition**
CR	Control Rod
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CRDH	Control Rod Drive Housing
CRDHS	Control Rod Drive Hydraulic System
CRGT	Control Rod Guide Tube
CRHA	Control Room Habitability Area
CRT	Cathode Ray Tube
CS&TS	Condensate Storage and Transfer System
CSDM	Cold Shutdown Margin
CS / CST	Condensate Storage Tank
CT	Main Cooling Tower
CTVCF	Constant Voltage Constant Frequency
CUF	Cumulative usage factor
CWS	Chilled Water System
D-RAP	Design Reliability Assurance Program
DAC	Design Acceptance Criteria
DAW	Dry Active Waste
DBA	Design Basis Accident
dc / DC	Direct Current
DCS	Drywell Cooling System
DCIS	Distributed Control and Information System
DEPSS	Drywell Equipment and Pipe Support Structure
DF	Decontamination Factor
D/F	Diaphragm Floor
DG	Diesel-Generator
DHR	Decay Heat Removal
DM&C	Digital Measurement and Control
DOF	Degree of freedom
DOI	Dedicated Operators Interface
DOT	Department of Transportation
dPT	Differential Pressure Transmitter
DPS	Diverse Protection System
DPV	Depressurization Valve
DR&T	Design Review and Testing
DTM	Digital Trip Module
DW	Drywell
EB	Electrical Building
EBAS	Emergency Breathing Air System
EBHV	Electrical Building HVAC

Term	Definition
ECCS	Emergency Core Cooling System
E-DCIS	Essential DCIS (Distributed Control and Information System)
EDO	Environmental Qualification Document
EFDS	Equipment and Floor Drainage System
EFPY	Effective full power years
EHC	Electrohydraulic Control (Pressure Regulator)
ENS	Emergency Notification System
EOC	Emergency Operations Center
EOC	End of Cycle
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedures
EPDS	Electric Power Distribution System
EPG	Emergency Procedure Guidelines
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
ERICP	Emergency Rod Insertion Control Panel
ERIP	Emergency Rod Insertion Panel
ESF	Engineered Safety Feature
ETS	Emergency Trip System
FAC	Flow-Accelerated Corrosion
FAPCS	Fuel and Auxiliary Pools Cooling System
FATT	Fracture Appearance Transition Temperature
FB	Fuel Building
FBHV	Fuel Building HVAC
FCI	Fuel-Coolant Interaction
FCM	File Control Module
FCS	Flammability Control System
FCU	Fan Cooling Unit
FDDI	Fiber Distributed Data Interface
FFT	Fast Fourier Transform
FFWTR	Final Feedwater Temperature Reduction
FHA	Fire Hazards Analysis
FIV	Flow-Induced Vibration
FMCRD	Fine Motion Control Rod Drive
FMEA	Failure Modes and Effects Analysis
FPS	Fire Protection System
FO	Diesel Fuel Oil Storage Tank
FOAKE	First-of-a-Kind Engineering
FPE	Fire Pump Enclosure
FTDC	Fault-Tolerant Digital Controller

Term	Definition
FTS	Fuel Transfer System
FW	Feedwater
FWCS	Feedwater Control System
FWS	Fire Water Storage Tank
GCS	Generator Cooling System
GDC	General Design Criteria
GDCS	Gravity-Driven Cooling System
GE	General Electric Company
GE-NE	GE Nuclear Energy
GEN	Main Generator System
GETAB	General Electric Thermal Analysis Basis
GL	Generic Letter
GM	Geiger-Mueller Counter
GM-B	Beta-Sensitive GM Detector
GSIC	Gamma-Sensitive Ion Chamber
GSOS	Generator Sealing Oil System
GWSR	Ganged Withdrawal Sequence Restriction
HAZ	Heat-Affected Zone
HCU	Hydraulic Control Unit
HCW	High Conductivity Waste
HDVS	Heater Drain and Vent System
HEI	Heat Exchange Institute
HELB	High Energy Line Break
HEP	Human error probability
HEPA	High Efficiency Particulate Air/Absolute
HFE	Human Factors Engineering
HFF	Hollow Fiber Filter
HGCS	Hydrogen Gas Cooling System
HIC	High Integrity Container
HID	High Intensity Discharge
HIS	Hydraulic Institute Standards
HM	Hot Machine Shop & Storage
HP	High Pressure
HPNSS	High Pressure Nitrogen Supply System
HPT	High-pressure turbine
HRA	Human Reliability Assessment
HSI	Human-System Interface
HSSS	Hardware/Software System Specification
HVAC	Heating, Ventilation and Air Conditioning
HVS	High Velocity Separator

Term	Definition
HWCS	Hydrogen Water Chemistry System
HWS	Hot Water System
HX	Heat Exchanger
I&C	Instrumentation and Control
I/O	Input/Output
IAS	Instrument Air System
IASCC	Irradiation Assisted Stress Corrosion Cracking
IBC	International Building Code
IC	Ion Chamber
IC	Isolation Condenser
ICD	Interface Control Diagram
ICS	Isolation Condenser System
IE	Inspection and Enforcement
IEB	Inspection and Enforcement Bulletin
IED	Instrument and Electrical Diagram
IEEE	Institute of Electrical and Electronic Engineers
IGSCC	Intergranular Stress Corrosion Cracking
IIS	Iron Injection System
ILRT	Integrated Leak Rate Test
IOP	Integrated Operating Procedure
IMC	Induction Motor Controller
IMCC	Induction Motor Controller Cabinet
IRM	Intermediate Range Monitor
ISA	Instrument Society of America
ISI	In-Service Inspection
ISLT	In-Service Leak Test
ISM	Independent Support Motion
ISMA	Independent Support Motion Response Spectrum Analysis
ISO	International Standards Organization
ITA	Inspections, Tests or Analyses
ITAAC	Inspections, Tests, Analyses and Acceptance Criteria
ITA	Initial Test Program
LAPP	Loss of Alternate Preferred Power
LCO	Limiting Conditions for Operation
LCW	Low Conductivity Waste
LD	Logic Diagram
LDA	Lay down Area
LD&IS	Leak Detection and Isolation System
LERF	Large early release frequency
LFCV	Low Flow Control Valve

Term	Definition
LHGR	Linear Heat Generation Rate
LLRT	Local Leak Rate Test
LMU	Local Multiplexer Unit
LO	Dirty/Clean Lube Oil Storage Tank
LOCA	Loss-of-Coolant-Accident
LOFW	Loss-of-feedwater
LOOP	Loss of Offsite Power
LOPP	Loss of Preferred Power
LP	Low Pressure
LPCI	Low Pressure Coolant Injection
LPCRD	Locking Piston Control Rod Drive
LPMS	Loose Parts Monitoring System
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
LWMS	Liquid Waste Management System
MAAP	Modular Accident Analysis Program
MAPLHGR	Maximum Average Planar Linear Head Generation Rate
MAPRAT	Maximum Average Planar Ratio
MBB	Motor Built-In Brake
MCC	Motor Control Center
MCES	Main Condenser Evacuation System
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MCRP	Main Control Room Panel
MELB	Moderate Energy Line Break
MLHGR	Maximum Linear Heat Generation Rate
MMI	Man-Machine Interface
MMIS	Man-Machine Interface Systems
MOV	Motor-Operated Valve
MPC	Maximum Permissible Concentration
MPL	Master Parts List
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSL	Main Steamline
MSLB	Main Steamline Break
MSLBA	Main Steamline Break Accident
MSR	Moisture Separator Reheater
MSV	Mean Square Voltage
MT	Main Transformer
MTTR	Mean Time To Repair

Term	Definition
MWS	Makeup Water System
NBR	Nuclear Boiler Rated
NBS	Nuclear Boiler System
NCIG	Nuclear Construction Issues Group
NDE	Nondestructive Examination
NE-DCIS	Non-Essential Distributed Control and Information System
NDRC	National Defense Research Committee
NDT	Nil Ductility Temperature
NFPA	National Fire Protection Association
NIST	National Institute of Standard Technology
NMS	Neutron Monitoring System
NOV	Nitrogen Operated Valve
NPHS	Normal Power Heat Sink
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRHX	Non-Regenerative Heat Exchanger
NS	Non-seismic
NSSS	Nuclear Steam Supply System
NT	Nitrogen Storage Tank
NTSP	Nominal Trip Setpoint
O&M	Operation and Maintenance
O-RAP	Operational Reliability Assurance Program
OBCV	Overboard Control Valve
OBE	Operating Basis Earthquake
OGS	Offgas System
OHLHS	Overhead Heavy Load Handling System
OIS	Oxygen Injection System
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLU	Output Logic Unit
OOS	Out-of-service
ORNL	Oak Ridge National Laboratory
OSC	Operational Support Center
OSHA	Occupational Safety and Health Administration
OSI	Open Systems Interconnect
P&ID	Piping and Instrumentation Diagram
PA/PL	Page/Party-Line
PABX	Private Automatic Branch (Telephone) Exchange
PAM	Post Accident Monitoring
PAR	Passive Autocatalytic Recombiner
PAS	Plant Automation System

Term	**Definition**
PASS	Post Accident Sampling Subsystem of Containment Monitoring System
PCC	Passive Containment Cooling
PCCS	Passive Containment Cooling System
PCT	Peak cladding temperature
PCV	Primary Containment Vessel
PFD	Process Flow Diagram
PGA	Peak Ground Acceleration
PGCS	Power Generation and Control Subsystem of Plant Automation System
PH	Pump House
PL	Parking Lot
PM	Preventive Maintenance
PMCS	Performance Monitoring and Control Subsystem of NE-DCIS
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PQCL	Product Quality Check List
PRA	Probabilistic Risk Assessment
PRMS	Process Radiation Monitoring System
PRNM	Power Range Neutron Monitoring
PS	Plant Stack
PSD	Power Spectra Density
PSS	Process Sampling System
PSWS	Plant Service Water System
PT	Pressure Transmitter
PWR	Pressurized Water Reactor
QA	Quality Assurance
RACS	Rod Action Control Subsystem
RAM	Reliability, Availability and Maintainability
RAPI	Rod Action and Position Information
RAT	Reserve Auxiliary Transformer
RB	Reactor Building
RBC	Rod Brake Controller
RBCC	Rod Brake Controller Cabinet
RBCWS	Reactor Building Chilled Water Subsystem
RBHV	Reactor Building HVAC
RBS	Rod Block Setpoint
RBV	Reactor Building Vibration
RC&IS	Rod Control and Information System
RCC	Remote Communication Cabinet
RCCV	Reinforced Concrete Containment Vessel
RCCWS	Reactor Component Cooling Water System

Term	**Definition**
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RDA	Rod Drop Accident
RDC	Resolver-to-Digital Converter
REPAVS	Refueling and Pool Area Ventilation Subsystem of Fuel Building HVAC
RFP	Reactor Feed Pump
RG	Regulatory Guide
RHR	residual heat removal (function)
RHX	Regenerative Heat Exchanger
RMS	Root Mean Square
RMS	Radiation Monitoring Subsystem
RMU	Remote Multiplexer Unit
RO	Reverse Osmosis
ROM	Read-only Memory
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRPS	Reference Rod Pull Sequence
RSM	Rod Server Module
RSPC	Rod Server Processing Channel
RSS	Remote Shutdown System
RSSM	Reed Switch Sensor Module
RSW	Reactor Shield Wall
RTIF	Reactor Trip and Isolation Function(s)
RT_{NDT}	Reference Temperature of Nil-Ductility Transition
RTP	Reactor Thermal Power
RW	Radwaste Building
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling
RWE	Rod Withdrawal Error
RWM	Rod Worth Minimizer
SA	Severe Accident
SAR	Safety Analysis Report
SB	Service Building
S/C	Digital Gamma-Sensitive GM Detector
SC	Suppression Chamber
S/D	Scintillation Detector
S/DRSRO	Single/Dual Rod Sequence Restriction Override
S/N	Signal-to-Noise
S/P	Suppression Pool
SAS	Service Air System
SB&PC	Steam Bypass and Pressure Control System

Term	Definition
SBO	Station Blackout
SBWR	Simplified Boiling Water Reactor
SCEW	System Component Evaluation Work
SCRRI	Selected Control Rod Run-in
SDC	Shutdown Cooling
SDM	Shutdown Margin
SDS	System Design Specification
SEOA	Sealed Emergency Operating Area
SER	Safety Evaluation Report
SF	Service Water Building
SFP	Spent fuel pool
SIL	Service Information Letter
SIT	Structural Integrity Test
SIU	Signal Interface Unit
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SMU	SSLC Multiplexing Unit
SOV	Solenoid Operated Valve
SP	Setpoint
SPC	Suppression Pool Cooling
SPDS	Safety Parameter Display System
SPTMS	Suppression Pool Temperature Monitoring Subsystem of Containment Monitoring System
SR	Surveillance Requirement
SRM	Source Range Monitor
SRNM	Startup Range Neutron Monitor
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SRS	Software Requirements Specification
SRSRO	Single Rod Sequence Restriction Override
SRSS	Sum of the squares
SRV	Safety Relief Valve
SRVDL	Safety relief valve discharge line
SSAR	Standard Safety Analysis Report
SSC(s)	Structure, System and Component(s)
SSE	Safe Shutdown Earthquake
SSLC	Safety System Logic and Control
SSPC	Steel Structures Painting Council
ST	Spare Transformer

Term	Definition
STP	Sewage Treatment Plant
STRAP	Scram Time Recording and Analysis Panel
STRP	Scram Time Recording Panel
SV	Safety Valve
SWH	Static water head
SWMS	Solid Waste Management System
SY	Switch Yard
TAF	Top of Active Fuel
TASS	Turbine Auxiliary Steam System
TB	Turbine Building
TBCE	Turbine Building Compartment Exhaust
TBE	Turbine Building Exhaust
TBLOE	Turbine Building Lube Oil Area Exhaust
TBS	Turbine Bypass System
TBHV	Turbine Building HVAC
TBV	Turbine Bypass Valve
TC	Training Center
TCCWS	Turbine Component Cooling Water System
TCS	Turbine Control System
TCV	Turbine Control Valve
TDH	Total Developed Head
TEMA	Tubular Exchanger Manufacturers' Association
TFSP	Turbine first stage pressure
TG	Turbine Generator
TGSS	Turbine Gland Seal System
THA	Time-history accelerograph
TLOS	Turbine Lubricating Oil System
TLU	Trip Logic Unit
TMI	Three Mile Island
TMSS	Turbine Main Steam System
TRM	Technical Requirements Manual
TS	Technical Specification(s)
TSC	Technical Support Center
TSI	Turbine Supervisory Instrument
TSV	Turbine Stop Valve
UBC	Uniform Building Code
UHS	ultimate heat sink
UL	Underwriter's Laboratories Inc.
UPS	Uninterruptible Power Supply
USE	Upper Shelf Energy

Term	Definition
USM	Uniform Support Motion
USMA	Uniform support motion response spectrum analysis
USNRC	United States Nuclear Regulatory Commission
USS	United States Standard
UV	Ultraviolet
V&V	Verification and Validation
Vac / VAC	Volts Alternating Current
Vdc / VDC	Volts Direct Current
VDU	Video Display Unit
VW	Vent Wall
VWO	Valves Wide Open
WD	Wash Down Bays
WH	Warehouse
WS	Water Storage
WT	Water Treatment
WW	Wetwell
XMFR	Transformer
ZPA	Zero period acceleration

5. REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 SUMMARY DESCRIPTION

The Reactor Coolant System (RCS) includes those systems and components that contain or transport fluids coming from or going to the reactor core. These systems form a major portion of the Reactor Coolant Pressure Boundary (RCPB). This chapter provides information regarding the reactor coolant system and pressure-containing appendages out to and including isolation valving. This grouping of components is defined as the RCPB.

The RCPB includes all pressure-retaining components such as pressure vessels, piping, pumps, and valves, which are:

- part of the RCS, or
- connected to the RCS up to and including any and all of the following:
  - the outermost containment isolation valve in piping that penetrates containment;
  - the second of the two valves normally closed during normal reactor operation in system piping that does not penetrate containment; and
  - the RCS safety/relief valve (SRV) piping and the depressurization valve piping.

This chapter also deals with various subsystems to the RCPB that are closely allied to it. Specifically, Section 5.4 describes these subsystems. The depressurization valve is part of the Automatic Depressurization System function of the Nuclear Boiler System discussed in Subsection 6.3.

The nuclear system pressure relief system protects the RCPB from damage due to overpressure. To protect against overpressure, pressure-operated safety/relief valves are provided that can discharge steam from the nuclear system to the suppression pool. The pressure relief system also acts to automatically depressurize the nuclear system in the event of a Loss-of-Coolant Accident (LOCA) in which the feedwater, Isolation Condenser (IC) and Control Rod Drive (CRD) System high pressure makeup fail to maintain reactor vessel water level. Depressurization of the nuclear system by actuation of the depressurization valves (DPVs) allows the Gravity-Driven Cooling System (GDCS) to supply cooling water to adequately cool the fuel.

Subsection 5.2.5 establishes the limits on nuclear system leakage inside the drywell so that appropriate action can be taken before the integrity of the nuclear system process barrier is impaired.

The reactor vessel and appurtenances are described in Section 5.3. The major safety consideration for the reactor vessel is the ability of the vessel to function as a radioactive material barrier. Various combinations of loading are considered in the vessel design. The vessel meets the requirements of applicable codes and criteria. The possibility of brittle fracture was considered, and suitable design, material selection, material surveillance activity, and operational limits were established that avoid conditions where brittle fracture was possible.

The RCS provides coolant flow through the core by natural circulation within the reactor vessel. The core coolant flow rate changes with reactor power output. The control rods are adjusted either manually or automatically with the Fine Motion Control Rod Drives (FMCRDs) to adjust reactor power. The natural recirculation within the reactor vessel eliminates the need for a recirculation system. Therefore, there are no large piping connections to the reactor vessel below the core and there are no recirculation pumps. The thermal-hydraulic design for reactor core coolant flow by natural recirculation is discussed in Section 4.4.

Main steamline flow restrictors of the venturi-type are part of the main steam nozzle on the reactor pressure vessel. The restrictors are designed to limit the loss of coolant resulting from a main steamline break inside or outside the containment. The restrictors limit the reactor depressurization rate to a value which will ensure that the steam dryer and other reactor internal structures remain in place and limit the radiological release outside of containment before closure of the Main Steam Isolation Valves (MSIVs).

Two isolation valves are installed on each main steamline. One is located inside the containment and the other is located outside the containment. If a main steamline break occurs inside the containment, closure of the isolation valve outside the containment seals the containment itself. The MSIVs automatically isolate the RCPB when a pipe break occurs outside containment. This action limits the loss of coolant and the release of radioactive materials from the nuclear system.

The CRD System provides makeup water via the Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) piping to the core anytime feedwater flow is not available. The system is started automatically upon receipt of a Level 2 reactor water level signal or manually by the operator. The CRD System is discussed in Section 4.6.

The RWCU/SDC System and the Isolation Condenser System (ICS) can be used to cool the nuclear system under a variety of situations. During normal shutdown and reactor servicing, the RWCU/SDC System removes residual and decay heat. The RWCU/SDC System in conjunction with the Isolation Condenser System (ICS) allows decay heat to be removed whenever the main heat sink (main condenser) is not available (e.g., hot standby). The ICS provides cooling of the reactor if the RCPB becomes isolated following a scram during power operations. The ICS automatically removes residual and decay heat to limit reactor pressure when reactor isolation occurs. Over a longer duration, the ICS provides a way to remove excess heat from the reactor with minimal loss of coolant inventory, if the normal heat removal path is unavailable.

The Gravity-Driven Cooling System (GDCS) is an engineered safety feature system for use during a postulated LOCA. The GDCS is operational at low reactor vessel pressure following pressure reduction by the Automatic Depressurization System (ADS) function of the Nuclear Boiler System. Operation of the GDCS and ADS is described in Section 6.3.

The RWCU/SDC System recirculates a portion of reactor coolant through a demineralizer to remove dissolved impurities with their associated corrosion and fission products from the reactor coolant. It also removes excess coolant from the reactor system under controlled conditions.

5.1.1 Schematic Flow Diagrams

Schematic flow diagrams (Figure 1.1-3a, Figure 1.1-3b and Figure 5.1-1) of the RCS show major components, principal pressures, temperatures, flow rates, and coolant volumes for normal steady-state operating conditions at rated power.

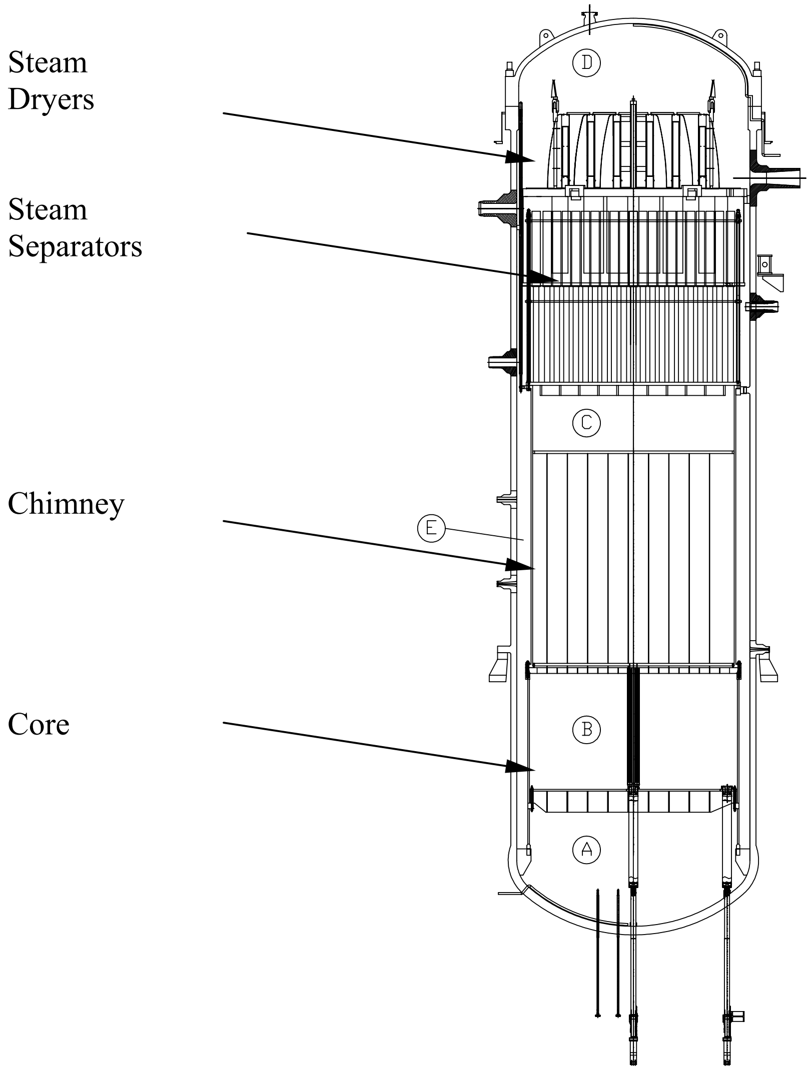
5.1.2 Piping and Instrumentation Schematics

Piping and instrumentation schematics covering the systems included within RCS and connected systems are presented as follows:

- Nuclear Boiler System (Figure 5.1-2);
- Isolation Condenser System (Figure 5.1-3);
- Reactor Water Cleanup/Shutdown Cooling System (Figure 5.1-4).

5.1.3 Elevation Schematics

The elevation schematic showing the principal features of the reactor and connecting systems in relation to the containment are provided in Figure 1.2-7, Figure 1.2-10 and Figure 1.2-11.



		Volume of Fluid (M^3)
A	Lower Plenum	103
B	Core	96
C	Upper Plenum (includes chimney and separator standpipe interior)	276
D	Dome (above normal water Level)	225
E	Downcomer Region	259

Figure 5.1-1. Coolant Volumes

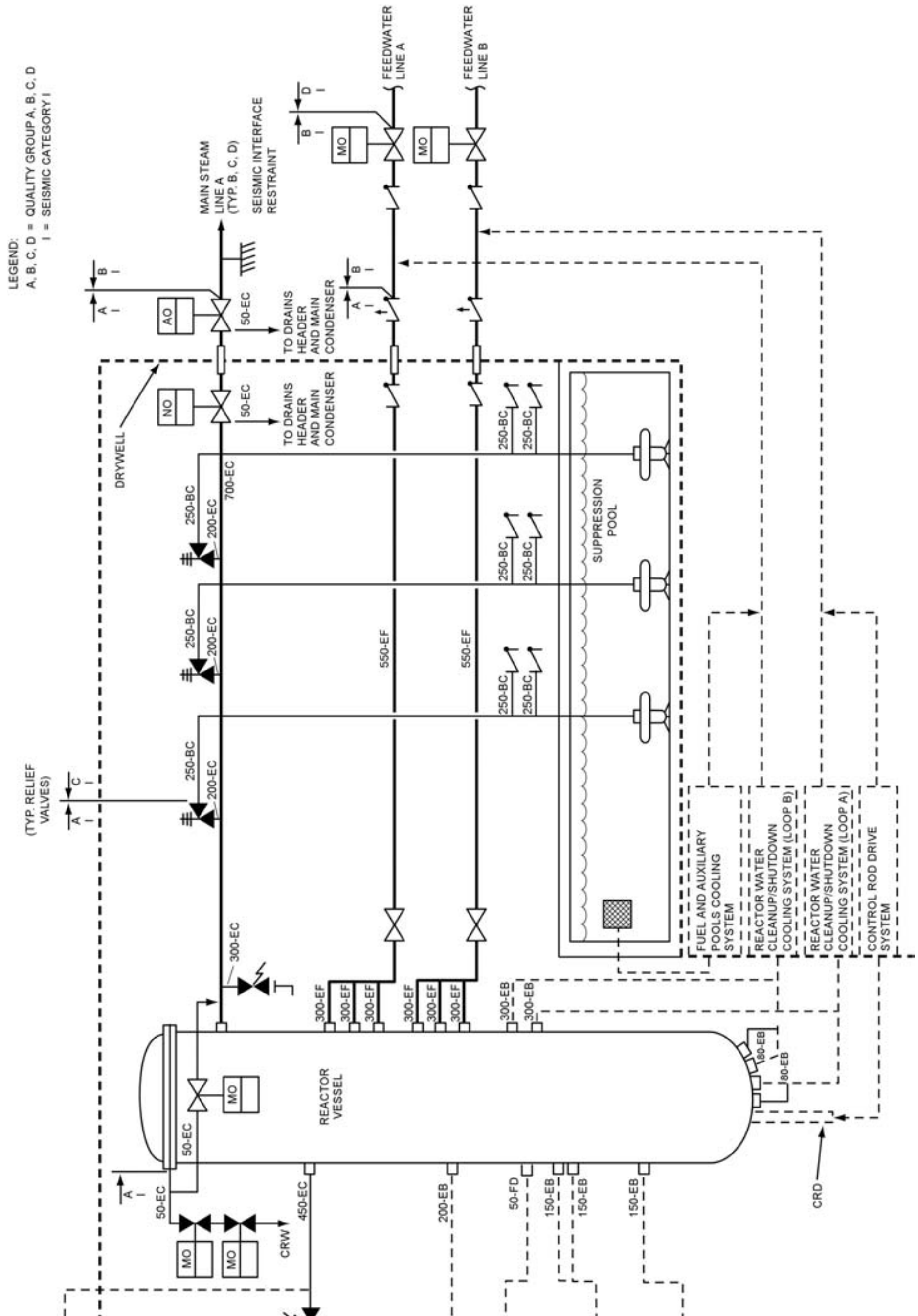


Figure 5.1-2. Nuclear Boiler System Schematic

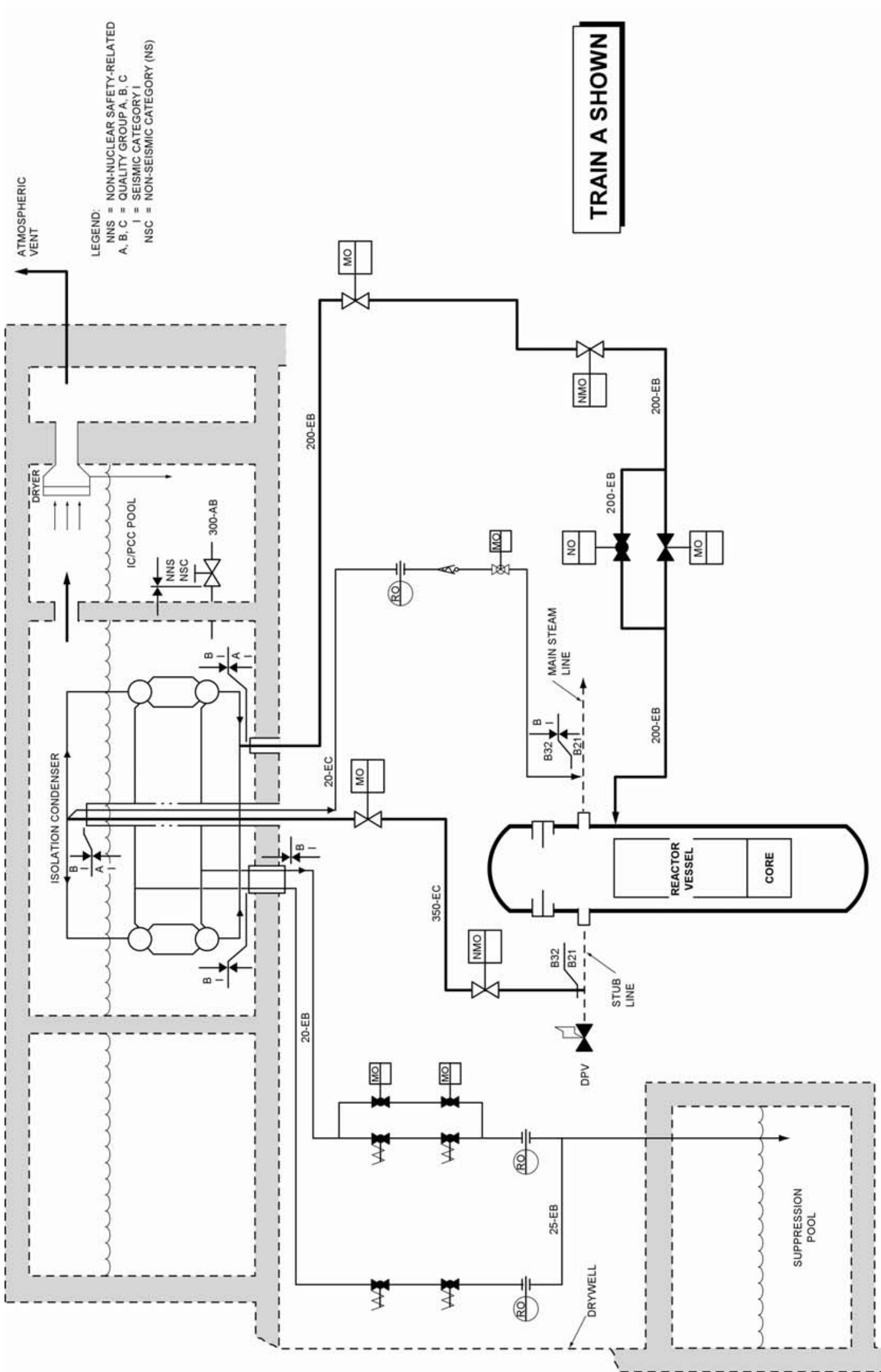


Figure 5.1-3. Isolation Condenser System Schematic

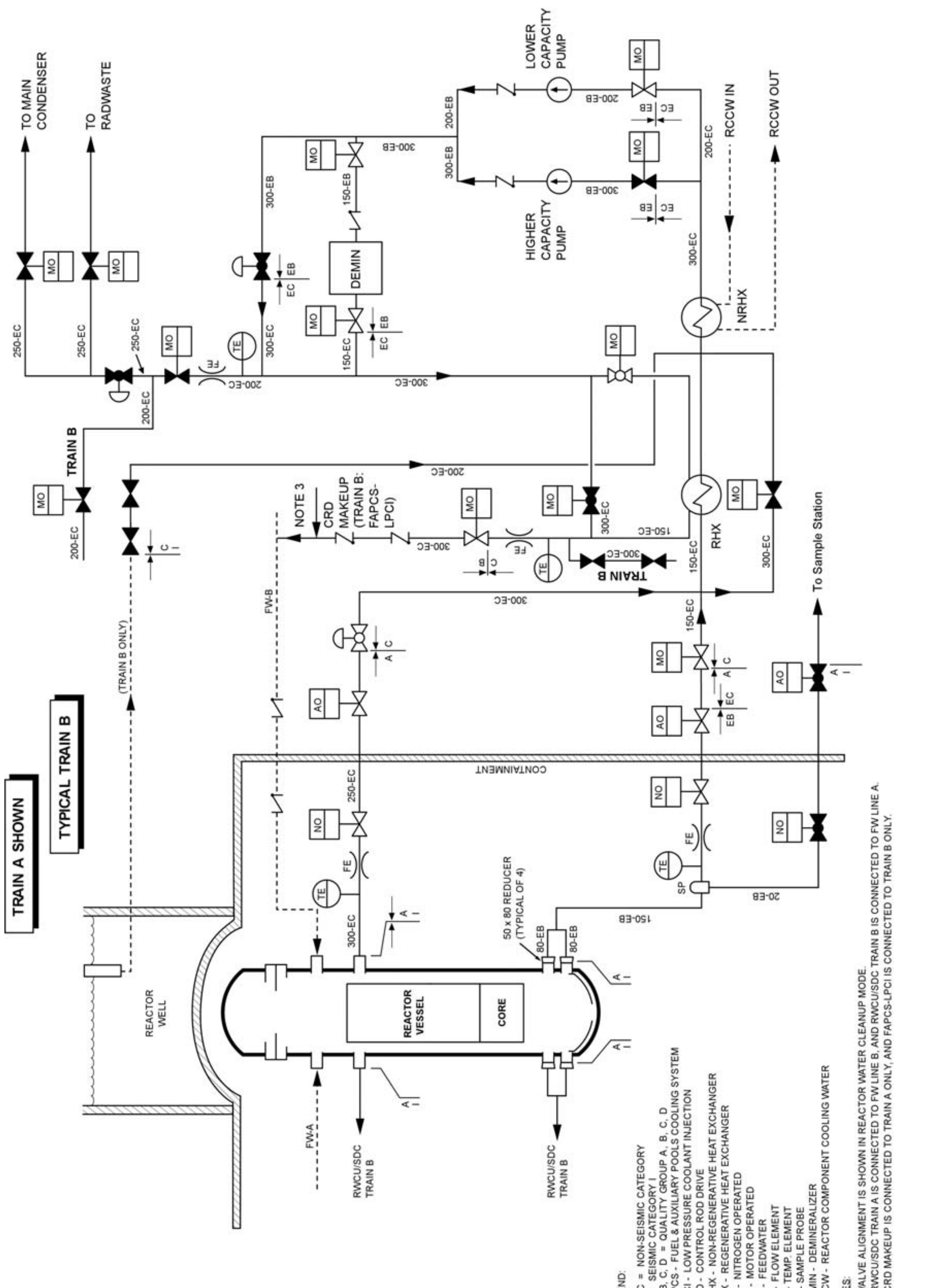


Figure 5.1-4. Reactor Water Cleanup/Shutdown Cooling System Schematic

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

This section discusses measures employed to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB).

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10 CFR 50, Section 50.55a

The ESBWR meets the relevant requirements of the following regulations:

(1) 10 CFR Part 50, Appendix A, General Design Criterion 1, as it relates to the requirement that safety-related structures, systems, and components are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

(2) 10 CFR Section 50.55a, as it relates to establishing minimum quality standards for the design, fabrication, erection, construction, testing and inspection of components within the reactor coolant pressure boundary and other safety-related fluid systems, by requiring conformance with appropriate editions of specified published industry codes and standards.

To meet the requirements of General Design Criterion 1 and 10 CFR Section 50.55a, Regulatory Guide 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," is used. This regulatory guide describes an acceptable method for determining quality standards for Quality Group B, C, and D water- and steam-containing safety-related components of water-cooled nuclear power plants.

Tables 3.2-1 and 3.2-3 show the Code applied to components. Code edition, applicable addenda, and component dates are in accordance with 10 CFR 50.55a.

5.2.1.2 Applicable Code Cases

The ESBWR meets the relevant requirements of the following regulations:

(1) 10 CFR Part 50, Appendix A, General Design Criterion 1, as it relates to the requirement that safety-related structures, systems and components are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

(2) 10 CFR Part 50, 50.55a, as it relates to the rule that establishes minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain components of boiling water reactor nuclear power plants by requiring conformance with appropriate editions of specified published industry codes and standards.

    To meet the requirements of General Design Criterion 1 and 10 CFR Part 50, 50.55a, the following regulatory guides are used:

    a. Regulatory Guide 1.84, “Code Case Acceptability in ASME Section III -Design and Fabrication.” This guide lists those Section III ASME Code Cases oriented to design and fabrication which are acceptable to the staff for implementation in the licensing of nuclear power plants.

b. Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1." This guide lists those Section XI ASME Code Cases which are acceptable to the staff for use in the inservice inspection of light-water-cooled nuclear power plants.

The reactor pressure vessel and appurtenances and the RCPB piping and valves are designed, fabricated, and tested in accordance with the applicable edition of the ASME Code, including addenda that were mandatory at the order date for the applicable components. Section 50.55a of 10 CFR 50 requires code case approval for Class 1, 2, and 3 components. These code cases contain requirements or special rules which may be used for the construction of pressure retaining components of Quality Group Classification A, B, and C. The various ASME Code cases that may be applied to components are listed in Table 5.2-1.

Regulatory Guides 1.84, and 1.147 provide a list of ASME Design and Fabrication code cases that have been generically approved by the Regulatory Staff. Code cases on this list may, for design purposes, be used until appropriately annulled. Annulled cases are considered active for equipment that has been contractually committed to fabrication prior to the annulment.

5.2.2 Overpressure Protection

This subsection evaluates systems that protect the RCPB from overpressurization.

As noted in SRP 5.2.2 Draft R3, overpressure protection for the reactor coolant pressure boundary (RCPB), during power operation of the reactor, is in compliance with ASME B&PV Code Section III and is ensured by application of relief and safety valves and the reactor protection system. For the ESBWR, the equipment includes Safety-Relief Valves (SRVs) on the main steam lines and piping from the SRVs to the suppression pool.

Overpressure protection for the RCPB, during low temperature operation of the plant (startup, shutdown), is ensured by the application of pressure relieving systems that function during the low temperature operation. For BWRs, no special area of review is required because BWRs never operate in water-solid conditions.

The ESBWR overpressure protection system meets the relevant requirements of the following regulations:

(1) General Design Criterion 15, as it relates to the reactor coolant system and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

The ESBWR meets the recommendations of the TMI action plan items of NUREG 0737 regarding testing to qualify reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents and the provision of direct indication of relief and safety valve position.

Other specific acceptance criteria of GDC 15 met by ESBWR are as follows:

For overpressure protection, the Isolation Condensers have sufficient capacity to preclude actuation of the SRVs, during normal operational transients, when assuming the following conditions at the plant:

a. The reactor is operating at licensed core thermal power level.

b. All system and core parameters are at values within normal operating range that produce the highest anticipated pressure.

c. All components, instrumentation, and controls function normally.

The SRVs have sufficient capacity to limit the pressure to less than 120% of the RCPB design pressure (as specified by the ASME Boiler and Pressure Vessel Code), during the most severe ATWS pressurization transient. Also, sufficient margin is available to account for uncertainties in the design and operation of the plant assuming:

(1) The reactor is operating at a power level that produces the most severe overpressurization event.

(2) All system and core parameters are at values within normal operating range, including uncertainties and technical specification limits that produce the highest anticipated pressure.

(3) The reactor scram is initiated by the second safety-grade signal from the reactor protection system.

(4) The discharge flow is based on the rated capacities specified in the ASME Boiler and Pressure Vessel Code, for each type of valve.

Full credit is taken for spring safety function of the SRVs designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code Section III. The spring-loaded valves are designed and constructed in accordance with ASME Code, Section III, NB 7640, as safety valves with auxiliary actuating devices.

5.2.2.1 Design Basis

Overpressure protection is provided in conformance with 10 CFR 50 Appendix A, General Design Criterion 15. Preoperational and startup instructions are given in Chapter 14.

Safety Design Bases

The nuclear pressure-relief system has been designed to:

- Prevent overpressurization of the nuclear system that could lead to the failure of the RCPB
- Provide automatic depressurization for breaks in the nuclear system so that the Gravity-Driven Cooling System (GDCS) can operate to protect the fuel barrier;
- Permit verification of operability; and withstand adverse combinations of loadings and forces resulting from normal, upset, emergency, or faulted conditions.

Power Generation Design Bases

The nuclear pressure-relief system SRVs have been designed to meet the following power generation bases:

- Discharge to the containment suppression pool; and
- Reclose following operation so that maximum operational continuity is obtained.

ASME Code

The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) requires that each vessel designed to meet Section III be protected from overpressure under Service Level B conditions.

The SRV setpoints are listed in Table 5.2-2 and satisfy the ASME Code specifications for safety valves.

The automatic depressurization capability of the nuclear system pressure relief system is evaluated in Section 6.3 and Subsection 7.3.1.1.

The following criteria are used in selection of SRVs:

- Must meet requirements of ASME Code Section III
- Must qualify for 100% of nameplate capacity at 103% of nameplate set pressure for the overpressure protection function; and
- Must meet other performance requirements necessary to provide safety and relief functions.

The SRV discharge piping is designed, installed, and tested in accordance with ASME Code Section III.

Safety/Relief Valve Capacity

SRV capacity is adequate to limit the primary system pressure, including transients, to the requirements of ASME Code Section III (Nuclear Power Plant Components), up to and including applicable addenda. The essential ASME requirements that are met by this analysis follow.

The rated capacity of the pressure-relieving devices shall be sufficient so that the rise in pressure within the protected vessel does not exceed 120% of the design pressure for pressurization events described in Chapter 15.

5.2.2.2 System Description

5.2.2.2.1 Piping and Instrument Diagrams

Figure 5.1-2 and Figure 5.2-1 show the schematic location of the pressure-relieving devices for:

- the reactor coolant system;
- the primary side of the auxiliary or emergency systems interconnected with the primary system; and
- any blowdown or heat dissipation system connected to the discharge side of the pressure-relieving devices.

The schematic arrangements of the SRVs are shown in Figures 5.2-1 through 5.2-3.

5.2.2.2.2 Equipment and Component Description

Description

The nuclear pressure-relief system consists of SRVs located on the main steamlines between the reactor vessel and the first isolation valve within the drywell. The SRVs are flange mounted onto forged outlet fittings located on the top of the main steamline piping in the drywell. Ten SRVs discharge through lines routed to quenchers in the suppression pool. The remaining eight SRVs are arranged into two groups of four. Each group discharges to a horizontal header that has a rupture disc at each end. Each header has a discharge line that is routed to a quencher in the suppression pool. These SRVs discharge through the rupture discs to the drywell or through the discharge line to the suppression pool. These valves protect against overpressure of the nuclear system and allow for manual or automatic reactor system depressurization.

The SRVs provide two main protection functions:

- overpressure safety operation (all eighteen of the valves are actuated by inlet steam pressure to prevent nuclear system overpressurization);
- depressurization operation [ten of the valves are actuated by the Automatic Depressurization System (ADS) as part of the Emergency Core Cooling System (ECCS) for events involving breaks in the nuclear system process barrier]

Chapter 15 discusses the events that are expected to activate the primary system SRVs. It also summarizes the number of valves expected to operate in safety (steam pressure) mode of operation during the initial blowdown of the valves and the expected duration of this first blowdown.

Remote manual actuation of the SRVs from the control room is recommended to minimize the total number of these discharges with the intent of achieving extended valve seat life.

Eight of the SRVs are opened by steam pressure initiated if the direct and increasing static inlet steam pressure overcomes the restraining spring and the frictional forces acting against the inlet steam pressure at the main or pilot disk and the main disk moves in the opening direction at a faster rate than corresponding disk movements at higher or lower inlet steam pressures. The condition at which this action is initiated is termed the “popping pressure” and corresponds to the set-pressure value stamped on the nameplate of the SRV.

Ten of the SRVs are opened by either of the following two modes of operation:

- The safety (steam pressure) mode of operation is initiated when the direct and increasing static inlet steam pressure overcomes the restraining spring and the frictional forces acting against the inlet steam pressure at the main or pilot disk and the main disk moves in the opening direction at a faster rate than corresponding disk movements at higher or lower inlet steam pressures. The condition at which this action is initiated is termed the “popping pressure” and corresponds to the set-pressure value stamped on the nameplate of the SRV.
- The ADS (power) mode of operation is initiated when an electrical signal is received at any of the solenoid valves located on the pneumatic actuator assembly. The solenoid valve(s) open, allowing pressurized nitrogen to enter the lower side of the pneumatic cylinder piston, which pushes the piston and the rod upwards. This action pulls the lifting mechanism of the main or pilot disk, thereby opening the valve to allow inlet steam to discharge through the SRV until the inlet pressure is near or equal to zero or the solenoid valve is closed.

The pneumatic operator is so arranged that, if it malfunctions, it does not prevent the valve from opening when steam inlet pressure reaches the spring lift set pressure.

For overpressure SRV operation (self-actuated or spring lift mode), the spring load establishes the safety valve opening setpoint pressure and is set to open at the setpoint designated in Table 5.2-2. In accordance with the ASME Code, the full lift of this mode of operation is attained at a pressure no greater than 3% above the setpoint. The opening time for the SRVs, from the time the pressure exceeds the valve set pressure to the time the valve is fully open, is less than 1.7 second.

The power-actuated SRVs can be operated individually by remote manual controls from the main control room.

The ADS utilizes ten of the SRVs for depressurization of the reactor as described in Section 6.3. Each of the SRVs is equipped with a pneumatic accumulator and check valve for the ADS and the manual opening functions. These accumulators assure that the valves can be opened following failure of the gas supply to the accumulators. The accumulator capacity is sufficient for one actuation at drywell design pressure.

Each ADS SRV discharges steam through a discharge line to a point below the minimum water level in the suppression pool water level. The discharge lines enter the suppression chamber below the suppression pool water level. The discharge lines are classified as Quality Group C and Seismic Category I.

Two vacuum relief valves are provided on each SRV discharge line to minimize initial rise of water in discharge piping and prevent drawing an excessive amount of water into the line as a result of steam condensation following termination of relief operation.

The ADS, which consists of SRVs and DPVs, automatically depressurizes the nuclear system sufficiently to permit the GDCS to operate. Further descriptions of the operation of the automatic depressurization feature are presented in Section 6.3.2.8.2 and within Subsection 7.3.1.

Design Parameters

The specified operating transients for components within the RCPB are presented in Section 3.9. Subsection 3.7.1 provides a discussion of the input criteria for design of Seismic Category I structures, systems, and components. The design requirements established to protect the principal components of the reactor coolant system against environmental effects are presented in Section 3.11.

Safety-Relief Valve

The design pressure and temperature of the valve inlet is 9.48 MPa gauge (1375 psig) at 307°C (585°F).

The valves have been designed to achieve the maximum practical number of actuations consistent with state-of-the-art technology.

5.2.2.2.3 Mounting of Safety/-Relief Valves

The SRVs are installed vertically on the main steam piping. The design criteria and analysis methods for considering SRV discharge loads are contained in Section 3.9.

5.2.2.2.4 Applicable Codes and Classification

The vessel overpressure protection system is designed to satisfy the requirements of Section III of the ASME Code. The general requirements for protection against overpressure of Section III of the Code recognize that reactor vessel overpressure protection is one function of the reactor protective systems and allows the integration of pressure-relief devices with the protective systems of the nuclear reactor. Hence, credit is taken for the scram protective system as a complementary pressure protection device. The NRC has also adopted the ASME Code as part of their requirements in the Code of Federal Regulations (10 CFR 50.55a).

5.2.2.2.5 Material Specifications

Material specifications for pressure-retaining components of SRVs are listed in Table 5.2-4.

5.2.2.3 Safety Evaluation

5.2.2.3.1 Method of Analysis

The method of analysis is approved by the United States Nuclear Regulatory Commission (NRC) or developed using criteria approved by the NRC

It is recognized that the protection of vessels in a nuclear power plant is dependent upon many protective systems to relieve or terminate pressure transients. Installation of pressure-relieving devices may not independently provide complete protection. The safety valve sizing evaluation gives credit for operation of the scram protective system which may be tripped by either one of three sources: (1) a direct valve position signal, (2) a flux signal, or (3) a high vessel pressure signal. The direct scram trip signal is derived from position switches mounted on the MSIVs. The position switches are actuated when the valves are closing and following 15% travel of full stroke. The flux signal is derived from the Neutron Monitoring System and is actuated at 125% of NBR. The pressure signal is derived from pressure transmitters piped to the vessel steam space.

Full account is taken of the pressure drop on both the inlet and discharge sides of the valves. All combination SRVs discharge into the suppression pool through a discharge pipe from each valve which is designed to achieve sonic flow conditions through the valve, thus providing flow independence to discharge piping losses.

5.2.2.3.2 System Design

A parametric study was conducted to determine the required steam flow capacity of the SRVs based on the following assumptions

Operating Conditions

- operating power = 4500 MWt (100% of nuclear boiler rated power);
- absolute vessel dome pressure ≤ 7.17 MPa (1040 psia); and
- steam flow = 2433 kg/s (19.31 Mlbm/hr).

These rated power conditions are the most severe because maximum stored energy exists at these conditions. At lower power conditions, the transients would be less severe.

Anticipated Operational Occurrences (AOOs)

The overpressure protection system is capable of accommodating the most severe pressurization transient. The ESBWR pressurization is mild relative to previous other BWR designs because of the large steam volume in the chimney and vessel head, which mitigates the pressurization. The scram and initial pressurization drops the water level below the feedwater sparger; when the feedwater system performs as expected, the spray of subcooled water condenses steam in the vessel steam space and immediately terminates the pressurization. For purposes of overpressure protection analyses, the feedwater system is assumed to trip at the initiation of the transient. Similarly, no credit is taken for the isolation condenser.

No credit is taken for the first scram signal that would occur (e.g., valve position for MSIV isolation). This is in accordance with NUREG-0800, Subsection 5.2.2, which requires that the reactor scram be initiated by the second safety-related signal from the Reactor Protection System (neutron flux for MSIV isolation, turbine trip and load rejection).

The evaluation of transient behavior based on the core loading shown in Figure 4.3-1 demonstrates that MSIV closure, with scram occurring on high flux, is the most severe pressurization transient, the result for this transient is similar to the TTNBPF. Other fuel designs and core loading patterns, including loading patterns similar to Figure 4.3-1, do not affect the conclusions of this evaluation. Analyses of this event will be performed for each core loading, and the results will be provided by the COL applicant to the NRC for information (Subsection 5.2.6). Table 5.2-3 lists the systems that could initiate during the MSIV closure event.

Evaluation Method

The evaluation method for overpressure protection transients is the TRACG computer code as described in Subsection 15.0.3.4.

SRV & AOO Analysis Specification

- Simulated valve groups:
  - spring-action safety mode (2 groups)
- Opening pressure setpoint (maximum safety limit):
  - spring-action safety mode (Table 5.2-2)
- Reclosure pressure setpoint (% of opening setpoint) both modes:
  - maximum safety limit (used in analysis): 96
  - minimum operational limit: 90

The opening and reclosure setpoints are assumed at a conservatively high level above the nominal setpoints. This is to account for initial setpoint errors and any instrument setpoint drift that might occur during operation. Conservative SRV response characteristics are also assumed; therefore, the analysis conservatively bounds all SRV operating conditions.

SRV Capacities

Sizings of the SRV capacities are based on establishing an adequate margin from the peak vessel bottom pressure to the vessel code limit in response to the reference AOOs.

The method used to determine total valve capacity is as follows.

Whenever the system pressure increases to the valve spring set pressure of a group of valves, these valves are assumed to begin opening and to reach full open at 103% of the valve spring set pressure.

5.2.2.3.3 Evaluation of Results

Total SRV Capacity

The required total SRV capacity is determined by analyzing the pressure rise from a MSIV closure transient with flux scram. Results of this analysis are given in Figure 5.2-4a through Figure 5.2-4f. The peak vessel bottom pressure calculated is below the acceptance limit of 9.481 MPa gauge (1375 psig). Figure 5.2-4a through Figure 5.2-4f show the MSIV isolation event with flux scram. The pressurization is not dynamic and does not significantly overshoot the relief valve setpoint ceases to increase once a single relief valve opens. Figure 5.2-4d shows that peak vessel pressure is only a function of the valve setpoint. This is because the higher steam volume-to-power ratio of the ESBWR causes the pressure rate prior to scram to be much lower than operating BWRs. After a scram, the pressure rates due to stored energy release are correspondingly lower. The peak pressure in these events is the relief valve setpoint, because, at these low pressurization rates, with large margin to the SRV setpoints, the pressure increase is effectively terminated by a relief flow of equivalent to 3 of the 18 valves.

Peak vessel bottom pressure is 8.71 Mpa gauge and peak dome pressure is 8.62 Mpa gauge at 38s.

Statistical Evaluation of MSIV closure event with flux scram

The statistical analysis performed to calculate the upper bound for the Maximum Vessel Pressure during MSIVF, perturbing the physical correlations and operating conditions (as explained in section 4 of reference 5.2-10), results in a maximum vessel pressure below 8.7 MPa with a 95/95 confidence. This is below the SRVs opening setpoint used in the analysis and is bounded by the calculation with no credit of Feed-water flow shown in Figure 5.2-4.

Pressure Drop in Inlet and Discharge

Pressure drop in the piping from the reactor vessel to the valves is taken into account in calculating the maximum vessel pressures. Pressure drop in the discharge piping to the suppression pool is limited by proper discharge line sizing to prevent backpressure on each SRV from exceeding 40% of the valve inlet pressure, thus assuring choked flow in the valve orifice and no reduction of valve capacity due to the discharge piping, for the 10 valves piped to the suppression pool or the discharge manifold for the 8 valves which discharge to the drywell.

5.2.2.3.4 System Reliability

The system is designed to satisfy the requirements of Section III of the ASME Code. The consequences of failure are discussed in Subsection 15.1.4 of this report.

5.2.2.4 Testing and Inspection Requirements

The inspection and testing of applicable SRVs utilizes a quality assurance program, which complies with Appendix B of 10 CFR 50.

The SRVs are tested at a suitable test facility in accordance with quality control procedures to detect defects and to prove operability prior to installation. The following tests are conducted:

- hydrostatic test at specified test conditions (ASME Code requirement based on design pressure and temperature);
- thermally stabilize the SRV to perform quantitative steam leakage testing at 1.03 MPaG (150 psig) below the SRV nameplate value with an acceptance criterion not to exceed 0.45 kg/hr (1 lbm/hr) leakage;
- full flow SRV test for set pressures and blowdown where the valve is pressurized with saturated steam, with the pressure rising to the valve set pressure (during production testing the SRV is adjusted to open at the nameplate set pressure ± 1%); and
- response time test where each SRV is tested to demonstrate acceptable response time based on system requirements. The valves are installed as received from the factory. The valve manufacturer certifies that design and performance requirements have been met. This includes capacity and blowdown requirements. The setpoints are adjusted, verified, and indicated on the valves by the vendor. Specified manual and automatic initiated signal for power actuation of each SRV is verified during the preoperational test program.
- It is not feasible to test the SRV setpoints while the valves are in place. The valves are mounted on flanges and can be removed for maintenance or bench testing and reinstalled during normal plant shutdowns. The valves will be tested in accordance with the requirements of the in-service testing program as discussed in Subsection 3.9.6 and Table 3.9-8 . The external and flange seating surfaces of all SRVs are 100% visually inspected when the valves are removed for maintenance or bench testing. Valve operability is verified during the preoperational test program as discussed in Chapter 14. As a part of the preoperational and startup testing of the main steamlines, movement of the SRV discharge lines will be monitored.

5.2.2.5 Instrumentation Requirements

None.

5.2.3 Reactor Coolant Pressure Boundary Materials

As discussed in SRP 5.2.3 Draft R3, this subsection addresses materials of the reactor coolant pressure boundary (RCPB) other than the reactor pressure vessel, which is covered in Subsection 5.3.1.

The ESBWR meets the requirements of 10 CFR Part 50 given below:

- General Design Criteria (GDC) 1 and 30, as they relate to quality standards for design, fabrication, erection and testing;
- GDC 4, as it relates to compatibility of components with environmental conditions;
- GDC 14 and 31, as they relate to extremely low probability of rapidly propagating fracture and gross rupture of the RCPB;
- Appendix B, Criterion XIII, as it relates to onsite material cleaning control;

- Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness of the RCPB; and
- Section 50.55a, as it relates to quality standards applicable to the RCPB.

5.2.3.1 Material Specifications

This subsection discusses the specifications for pressure-retaining ferritic materials, nonferrous metals and austenitic stainless steels, including weld materials, that are used for each component (e.g., vessels, piping, pumps, and valves) of the reactor coolant pressure boundary. The adequacy and suitability of the ferritic materials, stainless steels, and nonferrous metals specified for the above applications are also discussed.

Table 5.2-4 lists the principal pressure-retaining materials and the appropriate material specifications for the reactor coolant pressure boundary (RCPB) components; all RCPB materials conform to the American Society of Mechanical Engineers (ASME) Code, Section III.

5.2.3.2 Compatibility with Reactor Coolant

General corrosion and stress corrosion cracking induced by impurities in the reactor coolant can cause failures of the reactor coolant pressure boundary. The chemistry of the reactor coolant and any additives whose function is to control corrosion are reviewed in subsections 5.4.8, 9.3.9 and 9.3.10. The compatibility of the materials of construction employed in the RCPB with the reactor coolant, contaminants, or radiolytic products to which the system is exposed has been considered. The extent of the corrosion of ferritic low alloy steels and carbon steels in contact with the reactor coolant has been considered. Similarly, consideration has been given to uses of austenitic stainless steels in the sensitized condition. Special attention has been given to the use of austenitic stainless steels in any condition in BWRs considering the oxygen content of BWR coolant.

5.2.3.2.1 PWR Chemistry of Reactor Coolant

Not applicable to BWRs.

5.2.3.2.2 BWR Chemistry of Reactor Coolant

A brief review of the relationships between water chemistry variables and RCPB materials performance, fuel performance, and plant radiation fields is presented in this section. Further information may be obtained from Reference 5.2-3.

The major environment-related materials performance problem encountered to date in the RCPB of BWRs has been intergranular stress corrosion cracking (IGSCC) of sensitized austenitic stainless steel. IGSCC in sensitized material adjacent to welds in Type 304 and Type 316 stainless steel piping systems has occurred in the past. Substantial research and development programs have been undertaken to understand the IGSCC phenomenon and develop remedial measures. For the ESBWR, IGSCC resistance has been achieved through the use of IGSCC resistant materials such as Type 316 Nuclear Grade stainless steel and stabilized nickel-base niobium modified Alloy 600 and Alloy 82.

Much of the early remedy-development work focused on alternative materials or local stress reduction, but more recently the effects of water chemistry parameters on the IGSCC process have received increasing attention. Many important features of the relationship between BWR water chemistry and IGSCC of sensitized stainless steels have been identified.

Laboratory studies (References 5.2-1 and 5.2-2) have shown that, although IGSCC can occur in simulated BWR startup environments, most IGSCC damage probably occurs during power operation. The normal BWR environment during power operation is 286°C water containing dissolved oxygen, hydrogen and small concentrations of ionic and non-ionic impurities (conductivity generally below 0.3μS/cm at 25°C). It has been well documented that some ionic impurities (notably sulfate and chloride) aggravate IGSCC, and a number of studies have been made of the effect of individual impurity species on IGSCC initiation and growth rates (References 5.2-1 through 5.2-5). This work clearly shows that IGSCC can occur in water at 286°C with 200 ppb of dissolved oxygen, even at low conductivity (low impurity levels), but the rate of cracking decreases with decreasing impurity content. Although BWR water chemistry guidelines for reactor water cannot prevent IGSCC, maintaining the lowest practically achievable impurity levels will minimize its rate of progression (References 5.2-3 and 5.2-6).

Stress corrosion cracking of ductile materials in aqueous environments is often restricted to specific ranges of corrosion potential, so a number of studies of impurity effects on IGSCC have been made as a function of either corrosion potential or dissolved oxygen content. The dissolved oxygen content is the major chemical variable in BWR type water that can be used to manipulate the corrosion potential in laboratory tests (Reference 5.2-7).

In-reactor and laboratory evidence indicates that carbon and low alloy steels tend to show improved resistance to environmentally assisted cracking with both increasing water purity and decreasing corrosion potential (Reference 5.2-8).

Fuel Performance Considerations

Nuclear fuel is contained in Zircaloy tubes that constitute the first boundary or primary containment for the highly radioactive species generated by the fission process; therefore, the integrity of the tubes must be ensured. Zircaloy interacts with the coolant water and some coolant impurities. This results in oxidation by the water, increased hydrogen content in the Zircaloy (hydriding), and, often, buildup of a layer of crud on the outside of the tube. Excessive oxidation, hydriding, or crud deposition may lead to a breach of the cladding wall.

Metallic impurities can result in neutron losses and associated economic penalties which increase in proportion to the amount being introduced into the reactor and deposited on the fuel. With respect to iron oxide-type crud deposits, it can be concluded that operation within the BWR water chemistry guidelines provided in Table 5.2-5 (specifically the limits on feedwater iron levels) effectively precludes the buildup of significant deposits on fuel elements.

Radiation Field Buildup

The primary long-term source of radiation fields in most BWRs is Co^{60}, which is formed by neutron activation of Co^{59}. Corrosion products are released from corroding and wearing surfaces as soluble, colloidal, and particulate species. The formation of Co^{60} takes place after the corrosion products precipitate, adsorb, or deposit on the fuel rods. Subsequent re-entrainment in the coolant and deposition on out-of-core stainless steel surfaces leads to buildup of the activated corrosion products (such as Co^{60}) on the out-of-core surfaces. The deposition may occur either in a loosely adherent layer created by particle deposition, or in a tightly adherent corrosion layer incorporating radioisotopes during corrosion and subsequent ion exchange. Water chemistry influences all of these transport processes. The key variables are the concentration of soluble Co^{60} in the reactor water and the characteristics of surface oxides. Thus, any reduction in the soluble Co^{60} concentration has positive benefits.

As a means to reduce radiation field buildup, cobalt content has been reduced in alloys to be used in high fluence areas such as fuel assemblies and control rods. In addition, cobalt base alloys used for pins and rollers in control rods have been replaced with non-cobalt alloys. Furthermore, cobalt content is restricted in stainless steel components in the reactor vessel and other selected stainless steel components that have large surface areas exposed to high flow rates toward the reactor vessel.

The Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) system removes both dissolved and undissolved impurities, which can become radioactive deposits. Reduction of these radioactive deposits reduces occupational radiation exposure during operation and maintenance of the plant components.

Water quality parameters can have an influence on radiation buildup rates. In laboratory tests, the water conductivity and pH were varied systematically from a high purity base case. In each case, impurities increased the rate of Co^{60} uptake over that of the base case. The evidence suggests that these impurities change both the corrosion rate and the oxide film characteristics to adversely increase the Co^{60} uptake. Thus, controlling water purity should be beneficial in reducing radiation buildup.

Sources of Impurities

Various pathways exist for impurity ingress to the primary system. The most common sources of impurities that result in increases in reactor water conductivity are (1) condenser cooling water inleakage, (2) improper operation of ion exchange units, (3) air inleakage, and (4) radwaste recycle. In addition to situations of relatively continuous ingress, such as from low level condenser cooling water inleakage, transient events can also be significant. The major sources of impurities during such events are resin intrusions, organic chemical intrusions, inorganic chemical intrusions, and improper rinse of resins. Chemistry transients resulting from introduction of organic substances into the radwaste system comprise a significant fraction of the transients, which have occurred.

The condensate treatment system has two stages of water treatment. The first stage, high efficiency filters, is effective in removing insoluble solids, such as condensate system insoluble corrosion products. The second stage, the deep bed demineralizers, is effective in removing soluble solids, such as soluble corrosion products and impurities from condenser leakage.

The following factors are measured for control or diagnostic purposes to maintain proper water chemistry in the ESBWR:

Conductivity - Increasing levels of many ionic impurities adversely influence both the stress corrosion cracking behavior of RCPB materials, the rate of radiation field buildup, and may affect fuel performance. Therefore, conductivity levels in the reactor water are maintained at the lowest level practically achievable.

Chloride - Besides being promoters of IGSCC in sensitized stainless steels, chlorides are also capable of inducing transgranular cracking of nonsensitized stainless steels. Chlorides also promote pitting and crevice attack of most RCPB materials. Chlorides are normally associated with cooling water inleakage, but inputs via radwaste processing systems have also occurred.

Sulfate - Besides promoting IGSCC of sensitized Type 304 stainless steel in BWR-type water (in laboratory tests), sulfates have also been implicated in environment-assisted cracking of high-nickel alloys and carbon and low-alloy steels. Sulfate ingress can result from cooling water inleakage or resin ingress.

Oxygen - Besides being a major contributor to IGSCC of sensitized stainless steels, reduction of oxygen content is known to reduce the tendency for pitting and cracks of most plant materials.

During power operation, most of the oxygen content of reactor water is due to the radiolysis of water in the core and, therefore, oxygen control cannot be achieved through traditional chemistry and operational practices. Reactor water oxygen control to low, plant-specific levels can be obtained through hydrogen injection. Control of reactor water oxygen during startup/hot standby is accomplished by utilizing the deaeration capabilities of the condenser. Carbon steels exhibit minimal general corrosion and release rates in water with a conductivity less than 0.1μS/cm if the concentration of oxygen is in the range of 15 to 200 ppb.

Regulation of reactor feedwater dissolved oxygen to 30-200 ppb with a target of less than 100 ppb during power operation minimizes corrosion of the Condensate and Feedwater System and reduces the possibility of locally increasing reactor water oxygen concentrations. For oxygen concentrations below 15 ppb, the data indicates an increase in the corrosion and corrosion product release for carbon steels.

Iron - High iron inputs into the reactor are associated with excessive fuel deposit buildup. Proper regulation of feedwater purity and dissolved oxygen levels minimizes iron transport to the reactor. This, in turn, minimizes fuel deposits and assists in controlling radiation buildup.

Fluoride - Fluoride promotes many of the same corrosion phenomena as chloride (including IGSCC) and may also cause corrosion of Zircaloy core components. If fluoride is present, it is measured regularly for diagnostic purposes.

Organics - Organic compounds can be introduced into the RCPB via turbine or pump oil leakage, radwaste, or makeup water systems. Of particular concern is the possibility that halogenated organic compounds (e.g., cleaning solvents) may pass through the radwaste systems and enter the RCPB, where they decompose, releasing corrosive halogens (e.g., chlorides and fluorides).

Silica - Silica, an indicator of general system cleanliness, provides a valuable indication of the effectiveness of the RWCU/SDC System. Silica inputs are usually associated with incomplete silica removal in makeup water or radwaste systems.

pH - There are difficulties of measuring pH in low conductivity water. Nevertheless, pH of the liquid environment has been demonstrated to have an important influence on IGSCC initiation times for smooth stainless steel specimens in laboratory tests. In addition, pH can serve as a useful diagnostic parameter for interpreting severe water chemistry transients.

Electrochemical Corrosion Potential - The electrochemical corrosion potential (ECP) of a metal is the potential it attains when immersed in a water environment. The ECP is controlled by various oxidizing agents, including copper and radiolysis products. At low reactor water conductivities, the ECP of stainless steel should be controlled to suppress IGSCC.

Feedwater Hydrogen Addition Rate - A direct measurement of the feedwater hydrogen addition rate is made using the hydrogen addition system flow measurement device and is used to establish the plant-specific hydrogen flow requirements to satisfy the ECP limit of stainless steel. Subsequently, the addition rate measurements are used to help diagnose the origin of unexpected ECP changes.

Reactor Water Dissolved Hydrogen - A direct measurement of the dissolved hydrogen content in the reactor water serves as a cross check against the hydrogen gas flow meter in the injection system to confirm the actual presence and magnitude of the hydrogen addition rate.

Main Steamline Radiation Level - The major activity in the main steamline is N^{16} produced by a (n, p) reaction with O^{16} in the reactor water. Under conditions of HWC, the fraction of the N^{16} that volatilizes with the steam increases with increased dissolved hydrogen. The main steamline radiation monitor readings increase with the hydrogen addition rate. During initial plant testing, the amount of hydrogen addition required to reduce the electrochemical corrosion potential to the desired range is determined at various power levels.

The major impurities expected in the ESBWR under certain operating conditions are listed in Table 5.2-5.

To support water quality specifications flow-assisted corrosion resistant low alloy steels are to be used in susceptible steam extraction and drain lines. Stainless steels are considered for baffles, shields, or other areas of severe duty. Provisions are made to add nitrogen gas to extraction steamlines, feedwater heater shells, and drain piping to minimize corrosion during layup. Alternatively, the system may be designed to drain while hot so that dry layup can be achieved.

The potential deterioration of ESBWR carbon steel piping from flow-assisted corrosion due to high velocity single-phase water flow and two-phase steam/water flow is addressed by appropriate control of hydrodynamic and environmental conditions.

Water quality specifications for the ESBWR require the condenser to be designed and erected to minimize tube leakage and to facilitate maintenance. Condenser tubes and tube sheet are made of titanium or stainless steel alloys. Appropriate features are incorporated to detect leakage and segregate the source. The valves controlling the cooling water to the condenser sections are required to be operable from the control room so that a leaking section can be sealed off quickly.

Irradiation Assisted Stress Corrosion Cracking (IASCC) Considerations

Plant experience and laboratory tests indicate that irradiation assisted stress corrosion cracking (IASCC) can be initiated in solution annealed stainless steel above certain stress levels after exposure to radiation.

Extensive tests have also shown that IASCC has not occurred at fluence levels below $\sim 5 \times 10^{20}$ neutron/cm^2 ($E > 1$ MeV) even at high stress levels. Experiments indicate that as fluence increases above this threshold of 5×10^{20} neutron/cm^2, there is a decreasing threshold of sustained stress below which IASCC has not occurred.

Reactor core structural components are designed to be below these thresholds of exposure and/or stress to avoid IASCC. In addition, crevices have been eliminated from the top guide design in order to prevent the synergistic interaction with IASCC.

In areas where the $5x10^{20}$ neutron/cm^2 threshold of irradiation is not practically avoided, the stress level is maintained below the stress threshold. High purity grades of materials are used in control rods to extend their life. HWC, originally introduced on past BWR designs to control IGSCC, is also beneficial in avoiding IASCC.

5.2.3.2.3 Compatibility of Construction Materials with Reactor Coolant

The construction materials exposed to the reactor coolant consist of the following:

- solution-annealed austenitic stainless steels (both wrought and cast), Types 304, 304L, 316, 316L and XM-19;
- nickel-based alloy (including niobium modified Alloy 600 and X-750);
- carbon steel and low alloy steel;
- some 400-series martensitic stainless steel (all tempered at a minimum of 595°C);
- Colmonoy and Stellite hardfacing material (or equivalent); and
- precipitation hardening stainless steels, 17-4PH and XM-13 in the H1100 condition.

All of these construction materials are resistant to stress corrosion in the BWR coolant. General corrosion on all materials except carbon and low alloy steel is negligible. Conservative corrosion allowances are provided for all exposed surfaces of carbon and low alloy steels.

The requirements of GDC 4 relative to compatibility of components with environmental conditions are met by compliance with the applicable provisions of the ASME Code and by compliance with the recommendations of Regulatory Guide 1.44.

Contaminants in the reactor coolant are controlled to very low limits. These controls are implemented by limiting contaminant levels of elements (such as halogens, S, Pb) to as low as possible in miscellaneous materials used during fabrication and installation. These materials (such as tapes, penetrants) are completely removed and cleanliness is assured. Lubricant and gasket materials remain in contact with the coolant during operation and are evaluated on that basis. No detrimental effects will occur on any of the materials from allowable contaminant levels in the high purity reactor coolant. Expected radiolytic products in the BWR coolant have no adverse effects on the construction materials.

5.2.3.2.4 Compatibility of Construction Materials with External Insulation

All non-metallic insulation applied to austenitic stainless steel meets Regulatory Guide 1.36.

5.2.3.3 Fabrication and Processing of Ferritic Materials

This subsection discusses fracture toughness properties of ferritic materials used for pressure-retaining components of the RCPB, the control of welding in ferritic steels, and the requirements and methods for nondestructive examination of ferritic wrought seamless tubular products used for ASME Class 1 components of nuclear power plants as specified in the ASME Boiler and Pressure Vessel Code.

5.2.3.3.1 Fracture Toughness

All Class 1 carbon steel components are made from high toughness grade material.

In addition, all ferritic components shall comply with ASME Code requirements in accordance with the following:

- The ferritic materials used for piping, pumps, and valves of the reactor coolant pressure boundary are usually 63.5 mm (2.5 in.) or less in thickness. Impact testing is performed in accordance with NB-2332 for thicknesses of 63.5 mm (2.5 in.) or less. Impact testing is performed in accordance with ASME Code Paragraph NB-2331 for thickness greater than 63.5 mm.
- Materials for bolting with nominal diameters exceeding 25.4 mm (1 in.) are required to meet both the 0.64 mm (0.025 in.) lateral expansion specified in NB-2333 and the 61 J (45 ft.-lbf) Charpy V value. The 61 J (45 ft.-lbf) requirement of the ASME Code applies to bolts over 101.6 mm (4 in.) in diameter, however GE added the 61 J (45 ft.-lbf) as a more conservative requirement for nominal bolt diameters exceeding 25.4 mm (1 in.).
- The reactor vessel complies with the requirements of NB-2331 and 10CFR50 Appendix G. The reference temperature (RT_{NDT}) is established for all required pressure-retaining materials used in the construction of Class 1 vessels. This includes plates, forgings, weld material, and heat-affected zone. The RT_{NDT} differs from the nil-ductility temperature (NDT) in that, in addition to passing the drop weight test, three Charpy V-Notch specimens (transverse) must exhibit 68 J (50 ft.-lbf) absorbed energy and 0.89 mm (0.035 in.) lateral expansion at 33°C above the RT_{NDT}. The core beltline material must meet 104 J absorbed upper shelf energy. Consideration has been given to the effects of irradiation on beltline fracture toughness by controlling the chemical composition of vessel beltline materials.
- Calibration of instrument and equipment shall meet the requirements of the ASME Code, Section III, Paragraph NB/NC-2360.

5.2.3.3.2 Control of Welding

Regulatory Guide 1.50: Control of Preheat Temperature Employed for Welding of Low-Alloy Steel

Regulatory Guide 1.50 delineates preheat temperature control requirements and welding procedure qualifications supplementing those in ASME Sections III and IX.

Low-alloy steel is used only in reactor pressure vessel and feedwater piping. Other ferritic components in the reactor coolant pressure boundary are fabricated from carbon steel materials.

Preheat temperatures employed for welding of low alloy steel meet or exceed the recommendations of ASME Code Section III, Subsection NB. Components are either held for an extended time at preheat temperature to assure removal of hydrogen, or preheat is maintained until post-weld heat treatment. The minimum preheat and maximum interpass temperatures are specified and monitored.

All full penetration pressure-retaining welds are volumetrically examined.

Regulatory Guide 1.34: Control of Electroslag Weld Properties

Electroslag welding is not allowed on structural weld joints of low alloy steel.

Regulatory Guide 1.71: Welder Qualification for Areas of Limited Accessibility

Welder qualification for areas of limited accessibility is discussed under Regulatory Guide 1.71 in Subsection 5.2.3.4.2 of this report.

Moisture Control for Low Hydrogen, Covered Arc Welding Electrodes

Suitable identification, storage, and handling of electrodes, flux, and other welding material will be maintained. Precautions shall be taken to minimize absorption of moisture by electrodes and flux.

5.2.3.3.3 Nondestructive Examination of Tubular Products

Wrought tubular products that are used for pressure-retaining components of the RCPB are subject to the examination requirements of Paragraph NB-2550 of ASME Code Section III.

These RCPB components meet 10 CFR 50 Appendix B requirements and the ASME Code requirements, thus assuring adequate control of quality for the products.

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels

Austenitic stainless steels in a variety of product forms are used for construction of a limited number of pressure-retaining components in the reactor coolant pressure boundary. Process controls are exercised during various stages of component manufacturing and reactor construction to avoid severe sensitization of the material and to minimize exposure of the stainless steel to contaminants that could lead to stress corrosion cracking.

5.2.3.4.1 Avoidance of Stress/Corrosion Cracking

Avoidance of Significant Sensitization

When austenitic stainless steels are heated in the temperature range 427°C – 982°C, they are considered to become "sensitized" or susceptible to intergranular corrosion. The ESBWR design complies with Regulatory Guide 1.44 and with the guidelines of Generic Letter 88-01 and NUREG-0313 Revision 2, to avoid sensitization through the use of reduced carbon content and process controls.

All austenitic stainless steels are supplied in the solution heat treated condition and special sensitization tests are applied to confirm and assure proper heat treatment. For applications where stainless steel surfaces are exposed to reactor water at temperatures above 93°C in welded applications where solution heat treatment is not performed, nuclear grade materials (carbon content ≤ 0.02%) are used.

During fabrication, any heating operation (except welding) above 427°C is avoided, unless followed by solution heat treatment. During welding, heat input is controlled. The interpass temperature is also controlled. Where practical, shop welds are solution heat treated. In general, weld filler material used for austenitic stainless steel base metals is Type 308L/316L/309L/309MoL with an average ferrite content not less than 8FN (ferrite number).

Process Controls to Minimize Exposure to Contaminants

Process controls are exercised during all stages of component manufacturing and construction to minimize contaminants. Cleanliness controls are applied prior to any elevated temperature treatment. Exposure to contaminants capable of causing stress/corrosion cracking of austenitic stainless steel components are avoided by carefully controlling all cleaning and processing materials which contact the stainless steel during manufacture, construction, and installation.

Special care is exercised to insure removal of surface contaminants prior to any heating operations. Water quality for cleaning, rinsing, flushing, and testing is controlled and monitored. Suitable protective packaging is provided for components to maintain cleanliness during shipping and storage. The degree of surface cleanliness obtained by these procedures meets the requirements of Regulatory Guides 1.37 and 1.44.

Cold-Worked Austenitic Stainless Steels

Cold worked austenitic stainless steels are not used for RCPB components. Cold work controls are applied for components made of austenitic stainless steel. During fabrication, cold work is controlled by applying limits in hardness, bend radii and surface finish on ground surfaces.

5.2.3.4.2 Control of Welding

Avoidance of Hot Cracking

Regulatory Guide 1.31 describes the acceptable method of implementing requirements with regard to the control of welding when fabricating and joining austenitic stainless steel components and systems.

Written welding procedures that are approved by GE are required for all primary pressure boundary welds. These procedures comply with the requirements of Sections III and IX of the ASME Code and applicable NRC Regulatory Guides.

All austenitic stainless steel weld filler materials are required by specification to have a minimum delta ferrite content of 8 FN (ferrite number) and a maximum of 20 FN determined on undiluted weld pads by magnetic measuring instruments calibrated in accordance with AWS Specification A4.2.

Regulatory Guide 1.34: Electroslag Welds

See Regulatory Guide 1.34 in Subsection 5.2.3.3.2.

Regulatory Guide 1.71: Welder Qualification for Areas of Limited Accessibility

Regulatory Guide 1.71 requires that weld fabrication and repair for wrought low-alloy and high-alloy steels or other materials such as static and centrifugal castings and bimetallic joints comply with fabrication requirements of Sections III and IX of the ASME Code. It also requires additional performance qualifications for welding in areas of limited access.

All ASME Section III welds are fabricated in accordance with the requirements of Sections III and IX of the ASME Code. There are few restrictive welds involved in the fabrication of ESBWR components. Welder qualification for welds with the most restrictive access is accomplished by mockup welding. Mockups are examined by sectioning and radiography (or UT).

The Acceptance Criterion II.3.b.(3) of SRP Subsection 5.2.3 is based on Regulatory Guide 1.71. The ESBWR design meets the intent of this regulatory guide by utilizing the following alternate approach.

When access to a non-volumetrically examined ASME Section III production weld (1) is less than 300 mm in any direction and (2) allows welding from one access direction only, such weld and repairs to welds in wrought and cast low alloy steels, austenitic stainless steels and high nickel alloys and in any combination of these materials shall comply with the fabrication requirements specified in ASME Code Section III and with the requirements of Section IX invoked by Section III, supplemented by the following requirements:

- The welder performance qualification test assembly required by ASME Section IX shall be welded under simulated access conditions. An acceptable test assembly will provide a Section IX welder performance qualification required by this Regulatory Guide.
- If the test assembly weld is to be judged by bend tests, a test specimen shall be removed from the location least favorable for the welder. If this test specimen cannot be removed from a location prescribed by Section IX, an additional bend test specimen is required. If the test assembly weld is to be judged by radiography or UT, the length of the weld to be examined shall include the location least favorable for the welder.
- Records of the results obtained in welder accessibility qualification shall be as certified by the manufacturer or installer, shall be maintained and shall be made accessible to authorized personnel.
- Socket welds with a 50A nominal pipe size and under are excluded from the above requirements.
- For accessibility, when restricted access conditions obscure the welder's line of sight, the use of visual aids such as mirrors shall be used. The qualification test assembly shall be welded under the more restricted access conditions using the visual aid required for production welding.
- Surveillance of accessibility qualification requirements is performed along with normal surveillance of ASME Section IX performance qualification requirements.

5.2.3.4.3 Nondestructive Examination of Tubular Products

For discussion of nondestructive examination of tubular products, refer to Subsection 5.2.3.3.3.

5.2.4 Preservice and Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

This subsection describes the preservice and inservice inspection and system pressure test programs for NRC Quality Group A, AMSE. B&PV Code, Class 1 items. It describes these programs implementing the requirements of Subsection IWB of the ASME. B&PV Code Section XI.[1]

The design to perform preservice inspection is based on the requirements of the ASME Code, Section XI, 2001 Edition with 2003 Addenda. The development of the preservice and inservice inspection program plans is the responsibility of the Combined Operating License (COL) applicant and shall be based on the ASME Code, Section XI, Edition and Addenda specified in accordance with 10 CFR 50, Section 50.55a. The COL applicant is responsible for specifying

[1] Items as used in this subsection are products constructed under a certificate of authorization (NCA-3120) and material (NCA-1220). See Section III, NCA-1000, footnote 2.

the Edition of ASME Code Section XI to be used, based on the procurement date of the component per 10 CFR 50, Section 50.55a. The ASME Code requirements discussed in this section are provided for information and are based on the 2001 Edition of ASME Section XI with 2003 Addenda.

5.2.4.1 Class 1 System Boundary

Definition

The Class 1 system boundary for both preservice and inservice inspection programs and the system pressure test program includes all those items within the Class 1 and Quality Group A boundary on the piping and instrumentation schematics. Based on 10 CFR 50 (Effective Date 11-1-2004) and Regulatory Guide 1.26, Revision 3, the boundary includes the following:

- reactor pressure vessel;
- portions of the Main Steam System;
- portions of the Feedwater System;
- portions of the Standby Liquid Control System;
- portions of the Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) System;
- portions of the Isolation Condenser (IC) System; and
- portions of the Gravity-Driven Cooling (GDCS) System.

Those portions of the above systems within the Class 1 boundary are those items that are part of the RCS up to and including any and all of the following:

- the outermost containment isolation valve in the system piping which penetrates reactor containment;
- the second of two valves normally closed during normal reactor operation in system piping which does not penetrate reactor containment;
- the reactor coolant system SRVs and DPVs
- the Main Steam and Feedwater systems up to and including the outermost containment isolation valve.

Exclusions

Portions of the system within the reactor coolant pressure boundary (RCPB), as defined above, that are excluded from the Class 1 boundary in accordance with 10 CFR 50, Section 50.55a, are as follows:

- Those components where, in the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only; and
- Components that are or can be isolated from the reactor coolant system by two valves (both closed, both open, or one closed and one open). Each such open valve is capable of automatic actuation and, if the other valve is open, its closure time is such that, in the event of postulated failure of the component during normal reactor operation, each valve

remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

The description of portions of systems excluded from the RCPB does not address Class 1 components exempt from inservice examinations under ASME Code Section XI rules. The Class 1 components exempt from inservice examinations are described in ASME Code Section XI, IWB-1220.

5.2.4.2 Accessibility

All items within the Class 1 boundary are designed to provide access for the examinations required by ASME Section XI, IWB-2500. Responsibility for designing components for accessibility for preservice and inservice inspection is the responsibility of the COL holder. Items such as nozzle-to-vessel welds often may have inherent access restrictions when vessel internals are installed. Therefore, preservice examination shall be performed as necessary on these items prior to installation of internals which would interfere with examination.

Reactor Pressure Vessel Access

Access for examinations of the reactor pressure vessel (RPV) is incorporated into the design of the vessel, biological shield wall and vessel insulation as follows:

RPV Welds - The shield wall and vessel insulation behind the shield wall are spaced away from the RPV outside surface to provide access for remotely operated ultrasonic examination devices as described in Subsection 5.2.4.3. Access for the insertion of automated devices is provided through removable insulation panels and at shield wall hatches in the upper drywell area. Platforms are attached to the biological shield wall to provide access for installation of remotely operated examination devices.

RPV Head, RPV Studs, Nuts and Washers - The RPV head is dry stored on the refueling floor during refueling operations. Removable insulation is designed to provide access for manual ultrasonic examinations of RPV head welds. RPV nuts and washers are dry stored and are accessible for visual (VT-1) examination. RPV studs may be volumetrically examined in place or when removed.

Bottom Head Welds - Access to the bottom head to shell welds is provided from the lower drywell area through shield wall hatches and removable insulation panels around the cylindrical lower portion of the vessel. This design provides access for manual or automated ultrasonic examination equipment. Sufficient access is provided for partial penetration nozzle welds (i.e., CRD penetration and instrumentation nozzle welds) for performance of the visual VT-2 examination during the system leakage and system hydrostatic examinations.

Reactor Vessel Sliding Support - Access is provided for visual examination of the RPV Sliding Support per Subsection IWF. .

Piping, Pumps, Valves, and Supports - Physical arrangement of piping, pumps, and valves provide personnel access to each weld location for performance of ultrasonic and surface (magnetic particle or liquid penetrant) examinations and sufficient access to supports for performance of the visual VT-3 examination. Working platforms are provided in some areas to facilitate servicing of pumps and valves. Platforms and ladders are provided for access to piping welds, including the pipe-to-reactor vessel nozzle welds. Removable thermal insulation is provided on welds and components, which require frequent access for examination or are located in high radiation areas. Welds are located to permit ultrasonic examination from at least one side, but where component geometries permit, access from both sides is provided.

Restrictions: For piping systems and portions of piping systems subject to volumetric and surface examination, the following piping designs are not used:

- valve to valve;
- valve to reducer;
- valve to tee;
- elbow to elbow;
- elbow to tee;
- nozzle to elbow;
- reducer to elbow;
- tee to tee; and
- pump to valve.

Straight sections of pipe and spool pieces shall be added between fittings. The minimum length of the spool piece has been determined by using the formula L = 2T + 152 mm, where L equals the length of the spool piece (not including weld preparation) and T equals the pipe wall thickness.

5.2.4.3 Examination Categories and Methods

5.2.4.3.1 Examination Categories

The examination category of each item in accordance with ASME Section XI, IWB-2500 will be is listed in the preservice and inservice inspection programs prepared by the COL holder. The items will be listed by system and line number where applicable. The preservice and inservice inspection programs will also state the method of examination for each item.

For the preservice examination, all of the items selected for inservice examination shall be performed once in accordance with ASME Section XI, IWB-2200, with the exception of the examinations specifically excluded by ASME Section XI from preservice requirements, such as VT-3 examination of valve body and pump casing internal surfaces (B-L-2 and B-M-2 examinations categories, respectively) and the visual VT-2 examinations for categories B-E and B-P.

5.2.4.3.2 Examination Methods

Ultrasonic Examination of the Reactor Vessel

Ultrasonic examination for the RPV is conducted in accordance with the ASME Code, Section XI. The design to perform preservice inspection on the reactor vessel shall be based on the requirements of the ASME Code Section XI, 2001 Edition with 2003 Addenda. For the required preservice examinations, the reactor vessel shall meet the acceptance standards of Section XI, IWB-3510. The RPV shell welds are designed for 100% accessibility for both preservice and inservice inspection. RPV shell welds may be examined from the inside or outside diameter surfaces (or a combination of those techniques) using automated ultrasonic examination equipment. The RPV nozzle-to-shell welds will be 100% accessible for preservice inspection but might have limited areas that may not be accessible from the outer surface for inservice examination techniques; however, the inservice inspection program for the reactor vessel is the responsibility of the COL holder and any inservice inspection program relief request are reviewed by the NRC staff based on the Code Edition and Addenda in effect and inservice inspection techniques available at the time of COL application.

In most cases, inner radius examinations are performed from the outside of the nozzle using several compound angle transducer wedges to obtain complete coverage of the required examination volume.

Visual Examination

Visual examination methods VT-1, VT-2 and VT-3 shall be conducted in accordance with ASME Section XI, IWA-2210. In addition, VT-2 examinations shall meet the requirements of IWA-5240.

Where direct visual VT-1 examinations are conducted without the use of mirrors or with other viewing aids, clearance (of at least 610 mm of clear space) is provided where feasible for the head and shoulders of a man within a working arm's length (508 mm) of the surface to be examined.

At locations where leakages are normally expected and leakage collection systems are located (e.g., valve stems and pump seals), the visual VT-2 examination shall verify that the leakage collection system is operative.

Piping runs shall be clearly identified and laid out such that insulation damage, leaks and structural distress are evident to a trained visual examiner.

Surface Examination

Magnetic particle and liquid penetrant examination techniques shall be performed in accordance with ASME Section XI, IWA-2221 and IWA-2222, respectively. Direct examination access for magnetic particle (MT) and penetrant (PT) examination is the same as that required for direct visual (VT-1) examination (see Visual Examination), except that additional access shall be provided as necessary to enable physical contact with the item in order to perform the examination. Remote MT and PT generally are not appropriate as a standard examination process; however, boroscopes and mirrors can be used at close range to improve the angle of vision. As a minimum, insulation removal shall expose the area of each weld plus at least 152 mm from the toe of the weld on each side. Insulation is generally removed 406 mm on each side of the weld.

Volumetric Ultrasonic Direct Examination

Volumetric ultrasonic direct examination shall be performed in accordance with ASME Section XI, IWA-2232, which references mandatory Appendix I. In order to perform the examination, visual access to place the head and shoulders of the examiner within 508 mm of the area of interest shall be provided where feasible. Nine inches between adjacent pipes is sufficient spacing if there is free access on each side of the pipes. The transducer dimension has been considered: a 38 mm diameter cylinder, 76 mm long placed with access at a right angle to the surface to be examined. The ultrasonic examination instrument has been considered as a rectangular box 305 x 305 x 508 mm located within 12 m from the transducer. Space for a second examiner to monitor the instrument shall be provided if necessary.

Insulation removal for inspection is to allow sufficient room for the ultrasonic transducer to scan the examination area. A distance of 2T plus 152 mm, where T is pipe thickness, is the minimum required on each side of the examination area. The insulation design generally leaves 406 mm on each side of the weld, which exceeds minimum requirements.

Alternative Examination Techniques

As provided by ASME Section XI, IWA-2240, alternative examination methods, a combination of methods, or newly developed techniques may be substituted for the methods specified for a given item in this section, provided that they are demonstrated to be equivalent or superior to the specified method. This provision allows for the use of newly developed examination methods, techniques, etc., which may result in improvements in examination reliability and reductions in personnel exposure. IWA-2240 as written in the 1997 Addenda of ASME Section XI must be used when applying these provisions.

5.2.4.3.3 Data Recording

Manual data recording is performed where manual ultrasonic examinations are performed. Electronic data recording and comparison analysis are to be employed with automated ultrasonic examination equipment. Signals from each ultrasonic transducer are fed into a data acquisition system in which the key parameters of any reflectors are recorded. The data to be recorded for manual and automated methods are:

- location;
- position;
- depth below the scanning surface;
- length of the reflector;
- transducer data including angle and frequency; and
- calibration data.

The data so recorded shall be compared with the results of subsequent examinations to determine the behavior of the reflector.

5.2.4.3.4 Qualification of Personnel and Examination Systems for Ultrasonic Examination

Personnel performing examinations shall be qualified in accordance with ASME Section XI, Appendix VII. Ultrasonic examination systems shall be qualified in accordance with industry accepted programs for implementation of ASME Section XI, Appendix VIII. Qualification to ASME Section XI, Appendix VIII, in compliance with the provisions of 10CFR50, 50.55a shall be considered as a satisfactory alternative to Regulatory Guide 1.150.

5.2.4.4 Inspection Intervals

The inservice inspection intervals for the ESBWR conform to Inspection Program B as described in Section XI, IWB-2412. Except where deferral is permitted by Table IWB-2500-1, the percentages of examinations completed within each period of the interval shall correspond to Table IWB-2412-1 Inspection Program B provides for Inspection Intervals of a nominal length of 10 years with allowance for up to a year variation to coincide with refueling outages.

5.2.4.5 Evaluation of Examination Results

Examination results are evaluated in accordance with ASME Section XI, IWB-3000 with repairs based on the requirements of IWA-4000 and IWB-4000. Re-examination shall be conducted in accordance with the requirements of IWA-2200. The recorded results shall meet the acceptance standards specified in IWB-3400.

5.2.4.6 System Leakage and Hydrostatic Pressure Tests

System Leakage Tests

As required by Section XI, IWB-2500 for Category B-P, a system leakage test shall be performed in accordance with IWB-5200 on all Class 1 components and piping within the pressure-retaining boundary following each refueling outage. For the purposes of the system leakage test, the pressure-retaining boundary is defined in IWB-5222. The system leakage test shall include a VT-2 examination in accordance with IWA-5240. The system leakage test will be conducted at a pressure not less than that corresponding to 100% rated reactor power. The system hydrostatic test (described below), when performed, is acceptable in lieu of the system leakage test.

Hydrostatic Pressure Tests

A system hydrostatic test may be performed in lieu of a system leakage test, and when required for repairs, replacements, and modifications per IWA-4540. The test shall include all Class 2 or 3 pressure retaining components and piping within the boundaries defined by IWB-5230 or the boundary of a repair or replacement as applicable. The system hydrostatic test shall include a VT-2 examination in accordance with IWA-5240. For the purposes of determining the test pressure for the system hydrostatic test in accordance with IWB-5230, the nominal operating pressure shall be the maximum operating pressure indicated in the Process Flow Diagram (PFD) for the Nuclear Boiler System.

5.2.4.7 Code Exemptions

As provided in ASME Section XI, IWB-1220, certain portions of Class 1 systems are exempt from the volumetric and surface examination requirements of IWB-2500. Complete list will be provided in the plant-specific preservice inspection and inservice inspection program submitted by the Combined License applicant.

5.2.4.8 Code Cases

As applicable, the provisions of the Code Cases listed in Table 5.2-1 may be used for preservice and inservice inspections, evaluations, and repair and replacement activities.

5.2.5 Reactor Coolant Pressure Boundary (RCPB) Leakage Detection

As discussed in SRP 5.2.5, the Reactor Coolant Pressure Boundary (RCPB) leakage detection systems are designed to provide a means of detecting and, to the extent practical, identifying the source of the reactor coolant leakage. The system is designed to perform the detection and monitoring function to assure conformance with the requirements of General Design Criteria 2 and 30. The system design considers the following:

(1) The system is capable of identifying to the extent practical, the source of the reactor coolant leakage.

(2) The system is capable of separately monitoring and collecting leakage from both identifiable and unidentifiable sources.

(3) The system is adequately equipped with indicators and alarms for each leakage detection system in the main control room, and readily permits qualitative interpretations of such indicators.

(4) The system provides for the monitoring of systems connected to the RCPB for signs of intersystem leakage.

The design of the RCPB Leakage Detection Systems considers specific general design criteria and regulatory guides. The integrated design of the system is in accordance with the following criteria:

(5) General Design Criterion 2 as it relates to the capability of the systems to maintain and perform their safety functions following an earthquake, and meets the guidelines of Regulatory Guide 1.29, positions C-1 and C-2.

General Design Criterion 30 as it relates to the detection, identification and monitoring of the source of reactor coolant leakage, and meets the guidelines of Regulatory Guide 1.45, positions C-1 through C-9.

Leakage detection from the reactor coolant pressure boundary is the primary function of the Leak Detection and Isolation System (LD&IS). This system detects, monitors and alarms for leakage inside and outside the containment, and automatically initiates the appropriate protective action to isolate the source of the leak. The isolation function results in the closure of the appropriate containment inboard and outboard isolation valves to shut off leakage external to the containment. The system design for LD&IS control and instrumentation is described in Section 7.3.3. A simplified LD&IS system configuration is shown in Figure 7.3-3.

Abnormal leakages from various sources within the containment and from areas outside the containment are detected, monitored, alarmed and isolated as indicated in Table 5.2-6 and Table 5.2-7. In the event of a Loss-of-Coolant Accident (LOCA) that results in either high drywell pressure, or low reactor water level (Level 2), the isolation logic initiates closure of the containment isolation valves. As a backup to the Level 2 isolation logic, a discrete, hard-wired reactor water level (Level 1) logic is provided for containment isolation logic.

5.2.5.1 Leakage Detection Methods

The system is designed in conformance with Regulatory Guide 1.45 for leak-detection methods and functions, and with the applicable regulatory codes and standards that are listed for LD&IS in Table 7.1-1 for the isolation functions.

The leak detection methods that are employed inside and outside the containment are discussed and described separately in the following subsections.

5.2.5.1.1 Detection Methods of Leakage Within the Drywell

The primary detection methods that are used for monitoring small unidentified leakage are:

- the drywell floor drain high conductivity waste (HCW) sump pump activity;
- the drywell sump level changes;
- the drywell air coolers condensate flow rate; and
- the fission products radioactivity.

These parameters are continuously monitored and/or recorded in the main control room and alarmed on abnormal indications. The flow rate sensitivity for unidentified leakage in the drywell is 3.8 liter/min (1.0 gpm) within one hour.

The secondary detection methods used to detect gross unidentified leakage are the pressure and temperature parameters of the drywell atmosphere. High atmospheric pressure in the drywell trips the reactor and initiates isolation of the containment isolation valves. The ambient temperature in the drywell is also monitored and alarmed.

The detection of small identified leakage within the drywell is accomplished by monitoring the drywell equipment drain [Low Conductivity Waste (LCW)] sump pump activity and sump level increases. The sump instrumentation activates an alarm in the main control room when total leak flow rate exceeds 95 liters/min (25 gpm). Identified flow to the equipment drain sump is caused primarily from drainage emanating from the large process valves' stem packing seals.

Other leakage sources are also monitored and identified within the drywell, including (1) pressure leakage from the reactor vessel head flange seal, (2) drainage temperature from the valve stem packing seals, and (3) temperature in the SRV discharge lines to the suppression pool (monitored by Nuclear Boiler System). All of these leakage parameters are continuously monitored, recorded and alarmed in the main control room upon high indication levels.

Excessive leakage inside the drywell that could result from a process line break or LOCA, is detected by monitoring the drywell pressure and temperature for high indications, a low reactor water level, or a steamline high flow (for breaks downstream of the flow elements). The instrumentation channels for these monitored variables, except for the drywell temperature, trips the isolation logic upon abnormal indications and cause closure of the appropriate containment isolation valves.

The plant variables that are monitored for leakage detection within the primary containment are listed in Tables 5.2-8 and 5.2-9.

5.2.5.1.2 Detection of Leakage External to the Drywell

The areas outside the containment that are monitored for coolant leakage are:

- the equipment areas in the reactor building;
- the main steam tunnel area; and
- the turbine building.

Each area is instrumented to monitor the ambient temperature conditions and/or changes in differential temperatures that may be indicative of coolant leakage within its own boundary or compartment.

The temperature elements are located and shielded in such a manner so as to minimize sensor sensitivity to radiated heat from the piping or equipment. The trip setpoints are based on the room or compartment size and the cooling provisions of the ventilation system. The ambient temperature monitors initiate alarms in the main control room and trip the isolation logic to close the appropriate isolation valves. The differential temperature monitors are used only to initiate alarms to indicate small leakages.

Also, temperature elements are provided in the turbine building to monitor leakage from the steamlines to the turbine. These monitors initiate an alarm in the main control room and trip the isolation logic to close the MSIVs and the main steam drain line isolation valves on abnormal temperature.

Large leaks external to the containment are detected by indication of low reactor water level, high process line flow, high ambient temperatures, low steamline pressure or low main condenser vacuum. An abnormal indication from any of these monitored parameters initiate the appropriate alarm in the main control room and trip the isolation logic to cause closure of appropriate system isolation valves.

Intersystem radiation leakage into the Reactor Component Cooling Water System (RCCWS) from radioactive heat exchangers is monitored and alarmed by the Process Radiation Monitoring System (PRMS).

The variables monitored to detect leakage from piping and equipment located external to the primary containment are listed in Tables 5.2-8 and 5.2-9.

5.2.5.2 Leak Detection Instrumentation and Monitoring

5.2.5.2.1 Leak Detection Instrumentation and Monitoring Inside the Drywell

Drywell Floor Drain High Conductivity Waste (HCW) Sump Monitoring

The drywell floor drain sump collects unidentified leakage from such sources as floor drains, valve flanges, closed component cooling water for reactor equipment, condensate from the drywell air coolers and from any leakage not connected to the drywell equipment drain sump. The sump is equipped with two pumps and special monitoring instrumentation that measures the pump's operating frequency, the sump level and flow rates. These measurements are provided on a continuous basis to the main control room. The sump instrumentation is designed to detect reactor coolant leakage of 3.8 liters/min (1.0 gpm) within one hour and alarm at flow rates in excess of 19 liters/min (5 gpm).

Drywell Equipment Drain Low Conductivity Waste (LCW) Sump Monitoring

The drywell equipment drain sump collects only identified leakage from known sources such as the valve stem packings, RPV head flange seal, and from other known sources which are piped directly into the sump. This sump is equipped with two pumps and the same instrumentation as that used for the drywell floor drain sump. The same parameters are monitored and alarmed and the alarm setpoint has an adjustable range up to 95 liters/min (25 gpm).

Drywell Air Cooler Condensate Flow Monitoring

The condensate flow rate from the drywell air coolers is monitored for high drain flow, which could be indicative of leaks from piping or the equipment within the drywell. This flow is monitored by one instrumented channel using a bucket type flow transmitter located in the drywell. The flow measurement is provided to the main control room on a continuous basis for recording and alarming.

Drywell Temperature Monitoring

The ambient temperature within the drywell is monitored by four channels using temperature elements spaced equally in the vertical direction in the drywell. An abnormal increase in the drywell temperature could indicate a leak within the drywell, and would be alarmed in the main control room. These sensors are located such that they are sensitive to reactor coolant leakage and not to radiated heating from pipes and equipment.

Drywell Fission Product Monitoring

Primary coolant leaks and radioactivity within the drywell are detected through sampling and monitoring of the drywell atmosphere by the PRMS. The fission product monitor samples for radioactive particulates and radioactive noble gases. The radiation levels are recorded in the main control room and alarmed on abnormally high concentration levels.

Drywell Pressure Monitoring

The drywell pressure is monitored by four divisional channels using pressure transmitters to sense the drywell atmospheric pressure from four separate locations. A pressure rise above the nominal level indicates a possible leak or loss of reactor coolant within the drywell. A high pressure indication is alarmed in the main control room and initiates reactor trip and closure of the containment isolation valves.

Reactor Vessel Head Flange Seal Monitoring

A single pressure monitoring channel is provided for measurement of the pressure between the inner and outer reactor head flange seals. A high pressure indicates a leak in the inner O-ring seal. This pressure is monitored and is annunciated in the main control room upon high level indication. Leakage from the reactor head flange is directed to the drywell equipment drain sump.

Isolation Condenser Steamline and Condensate Return Line Flow Monitoring

The steamline flow to each isolation condenser is monitored by four divisional channels using differential pressure transmitters to sense the pressure difference across elbow tabs located in the main steamline to the condenser. The condensate flow from the condenser back to the vessel is monitored by similar instrumentation. A high flow rate in either line could indicate a leaking isolation condenser or a line break. A high flow indication is alarmed in the main control room and initiates closure of the isolation condenser isolation valves.

Safety/Relief Valve (SRV) Leakage Monitoring

Leakage from each SRV is monitored by a single channel using a temperature element to detect for steam discharge. Each temperature channel initiates a common alarm in the main control room upon high temperature indication in any of the SRV discharge lines. The temperature sensors are mounted in thermowells in the discharge piping located several feet from the valve body to prevent false indication. SRV leakage monitoring is provided by the Nuclear Boiler System.

Valve Stem Packing Leakage Monitoring

Leakage monitoring is provided for the remote power-operated valves in the containment that are fitted with double valve stem packing with leak-off lines between the packing. Leakage through the inner packing is carried to the drywell equipment drain sump and is monitored by a temperature sensor. A remote operated solenoid valve is provided to isolate the leakage flow.

Main Steamline High Flow Monitoring

The flow in each main steamline is monitored by four divisional channels using differential pressure transmitters to sense the pressure difference across a flow restrictor in the line. A high flow rate in the main steamline could indicate a break in one or more of the lines downstream from the flow restrictors. A high flow in any of the main steamlines is annunciated in the main control room, resulting in isolation of all MSIVs and main steam drain valves.

Reactor Vessel Low Water Level Monitoring

The Nuclear Boiler System provides two sets of four divisional channels to the LD&IS, one set each for low reactor water level L1 and L2, for containment isolation. Each level measurement is monitored by four level transmitters, and low level is annunciated in the main control room. Reactor water level L1 is provided as a backup to L2 for reliability to ensure containment isolation.

Reactor Well Liner Leakage Monitoring

Leakage from the reactor well liner and from the bellows seal is monitored visually.

5.2.5.2.2 Leak Detection Instrumentation and Monitoring External to Drywell

Visual Inspection of Accessible Plant Areas

Accessible areas are inspected periodically and the operability of the leak detection instrumentation is verified regularly. Any abnormal leakage that is detected will be investigated for corrective action.

Reactor Building Floor and Equipment Drain Sump Monitoring

In the reactor building, the equipment drain sumps collect the identified leakage from known sources in enclosed equipment areas. Leakage from unknown sources, such as the RWCU/SDC system lines, process instrument piping, etc. is collected in the floor drain sumps. The number of pumps and the instrumentation used by the reactor building floor and equipment drain sumps are similar to those provided for the drywell drain sumps (Subsection 5.2.5.2.1). The sump levels and the pump operating frequency are monitored. Alarms are activated in the main control room when setpoints are exceeded.

Reactor Water Cleanup/Shutdown Cooling System Flow Monitoring

The mass flow rate in each of the RWCU/SDC system train piping inside and outside the containment is measured by venturi type flow elements and transmitters and temperature elements (for density correction) in each of the four divisions of LD&IS. A high differential mass flow rate between inside and outside the containment is indicative of leakage within the RWCU/SDC train or a line break. Both the inboard and outboard containment isolation valves of the affected train are isolated and an alarm is activated in the main control room.

Main Steamline Tunnel Area Temperature Monitoring

In the reactor building, the ambient air temperature in the main steamline tunnel area is monitored by four divisional channels using thermocouple temperature elements. A high ambient temperature within the tunnel area is annunciated in the main control room and initiates isolation of the main steamlines and the RWCU/SDC process lines. In addition to leakage from the main steamlines, a high ambient temperature in the main steamline tunnel area can also indicate leakage from the reactor feedwater or RWCU/SDC piping. Isolation of the feedwater lines, if necessary, can be accomplished manually by the operator.

In the turbine building, the ambient air temperature in the steamline area is monitored by four divisional channels using thermocouples located at different places along the steamline. A high ambient temperature is annunciated in the main control room and initiates isolation of the steamlines to the turbine.

All thermocouples are located away from the main steamlines and are shielded to be only sensitive to ambient air temperatures and not to the radiated heat from the steamlines. Isolation of the main steamlines is accomplished through simultaneous closure of all the MSIVs and the steam drain line valves.

Isolation Condenser Radiation Leakage Monitoring

The vent discharge from each isolation condenser into the pool area is monitored separately for high radiation levels by the PRMS. Four divisional channels per isolation condenser are provided to sense for gamma radiation leakage using digital gamma sensitive detectors. A high radiation level is annunciated in the main control room and causes isolation of the defective isolation condenser.

Main Steamline Low Pressure Monitoring

The main steamline flow is monitored for low pressure by four pressure transmitters (one in each line) that sense the pressure downstream of the outboard MSIVs. The sensing points are located as close as possible to the turbine stop valves. A low steamline pressure can be an indication of a steamline leak or a malfunction of the reactor pressure control system. The isolation logic automatically initiates closure of all MSIVs and the main steamline drain valves if pressure at the turbine falls below the setpoint during reactor operation.

Main Condenser Low Vacuum Monitoring

The pressure in the main condenser is monitored for low vacuum to prevent overpressure of condenser upon loss of vacuum. Four divisional pressure monitoring channels are provided to generate the trip on low vacuum level. The trip signal is used by the isolation logic for closure of the MSIVs and the steam drain line valves. The condenser vacuum measurement is bypassed during startup and shutdown operations to guard against unnecessary isolation.

Intersystem Leakage Monitoring

Intersystem leakage of radioactive material into each RCCWS train is monitored continuously by the PRMS. A radiation monitor is provided at the RCCWS common discharge line that connects the cooling water output flows from the RWCU/SDC non-regenerative heat exchanger, the FAPCS heat exchanger, the upper and lower drywell coolers, the reactor building chiller, and the RCCWS air cooler. A high level of radioactivity is indicative of reactor coolant leakage into the closed loop RCCWS train. The high radiation level is alarmed in the control room.

Differential Temperature Monitoring in Equipment Areas

Differential temperature monitoring is provided in key areas in the reactor building to detect for small leaks. Such areas as the main steamline tunnel and the equipment areas of the RWCU/SDC System are instrumented with thermocouples that provide differential temperature measurements for alarm indication only.

Large Leaks External to the Drywell

The instrumentation provided to monitor main steamline flow, reactor vessel low water levels, IC steamline flow, and RWCU/SDC reactor coolant flow (as discussed under the appropriate paragraphs in Subsections 5.2.5.2.1 and 5.2.5.2.2) also indicates large leaks from the reactor coolant piping external to the drywell.

5.2.5.2.3 Summary of Plant Variables Monitored for Leak Detection

The plant variables monitored for leakage are summarized in Tables 5.2-8 and 5.2-9 for areas within and outside the containment. The automatic LD&IS isolation functions that are provided for detection and isolation of gross leakage within the plant are identified in Table 5.2-6. The leakage parameters of the plant that are monitored and annunciated in the main control room are identified in Table 5.2-7. Also, Table 5.2-6 lists at least two or more leakage parameters that are monitored for containment isolation.

5.2.5.3 Display and Indications in the Main Control Room

Monitored plant leakage parameters are measured, recorded and displayed on the appropriate panels in the main control room. All abnormal indications are annunciated to alert the operator for corrective action. All initiated isolation functions are also alarmed in the main control room.

5.2.5.4 Limits for Reactor Coolant Leakage Rates Within the Drywell

The total reactor coolant leakage rate consists of all identified and unidentified leakages that flow to the drywell floor drain and equipment drain sumps. The reactor coolant leakage rate limits for alarm annunciation are established at less than or equal to 95 liters/min (25 gpm) from identified sources and at 19 liters/min (5 gpm) from unidentified sources. The instrumentation is designed to measure leakage rates from unidentified sources of 3.8 liters/min (1 gpm) in one hour.

5.2.5.5 Criteria to Evaluate the Adequacy and Margin of Leak Detection System

For process lines that penetrate the containment, at least two different methods are used for detecting and isolating the leakage for the affected system. The instrumentation is designed to initiate alarms at established leakage limits and isolate the affected systems. The alarm setpoints are determined analytically or are based on actual measurements made during startup and pre-operational tests.

The unidentified leakage rate limit is based, with an adequate margin for contingencies, on a crack size large enough for leakage to propagate rapidly. The established limit is sufficiently low so that, even if the entire leakage rate were coming from a single crack in the nuclear system process barrier, corrective action could be taken before barrier integrity is threatened.

The sumps instrumentation is capable of detecting unidentified leakage of 3.8 liters/min (1 gpm) in one hour within the drywell.

5.2.5.6 Separation of Identified and Unidentified Leakages in the Containment

Identified and unidentified leakages from sources within the drywell are collected and directed to separate sumps, the LCW equipment drain sumps for identified leakages and the HCW floor drain sumps for unidentified leakages.

5.2.5.7 Testing, Calibration and Inspection Requirements

The requirements for testing, calibration and inspection of the LD&IS are covered in Subsection 7.3.3.4.

5.2.5.8 Regulatory Guide 1.45 Compliance

This regulatory guide specifies acceptable leak detection methods and flow rate limits for use in monitoring and detecting leaks from the reactor coolant pressure boundary.

Leakage is collected separately in drain sumps from identified and unidentified sources in the containment and total flow rate from each sump is independently monitored, thus satisfying Regulatory Guide 1.45, Position C.1.

Leakage from unidentified sources from inside the drywell is collected into the floor drain sump to detect leakage of 3.8 liters/min (1 gpm), thus satisfying Regulatory Guide 1.45, Position C.2.

Three separate detection methods are used for leakage monitoring: (1) the floor drain sump level and pump operating frequency, (2) radioactivity of the airborne particulates, and (3) the drywell air coolers condensate flow rate, thus satisfying Regulatory Guide (RG) 1.45, Position C.3.

Intersystem radiation leakage into the Reactor Component Cooling Water System is monitored as described in Subsection 5.2.5.2.2, thus satisfying RG 1.45, Position C.4.

The monitoring instrumentation of the drywell floor drain sump, the air particulate radioactivity, and the drywell air cooler condensate flow rate are designed to detect leakage rates of 3.8 liters/min (1 gpm) within one hour, thus satisfying RG 1.45, Position C.5.

The leak detection system required to perform isolation functions is classified Class 1E, Seismic Category I; and the system is designed to operate during and following seismic events, thus meeting the intent of 10 CFR 50, Appendix A, GDC 2. The airborne particulate radioactivity monitor is designed to operate during an SSE event. Thus, RG 1.45, Position C6 is satisfied.

Each monitored leakage parameter is indicated in the main control room and activates an alarm on abnormal indication. Procedures are provided to the operator to convert the identified and unidentified leakages into a common leakage rate equivalent to determine that the total leakage rate is within the technical specification limit. Each monitored leakage channel of the LD&IS can be tested and calibrated separately during normal plant operation without causing a plant outage. This information satisfies RG 1.45, Position C.7.

The LD&IS sensors and channels are periodically tested and calibrated during reactor operation, thus satisfying RG 1.45, Position C.8.

The following methods are used to verify operability:

- simulation of signals to initiate trips;
- channel-to-channel comparison of the same monitored leakage parameter;
- operability checks by comparing one method with another; and
- continuous monitoring of leakage parameters.

The limits established for alarming unidentified and identified leakages are less than or equal to 19 liters/min (5 gpm) and 95 liters/min (25 gpm), respectively. This satisfies Position C.9 of RG 1.45.

5.2.6 COL Information

Overpressure Protection

The COL applicant is required to submit an overpressure protection analysis for core loadings different than the reference ESBWR core loading.

Preservice and Inservice Inspection Program Plan

The COL holder is responsibility for the development of the preservice and inservice inspection program plans that are based on the ASME Code, Section XI. The COL applicant is responsible for specifying the Edition of ASME Code Section XI to be used.

5.2.7 References

5.2-1 D. A. Hale, "The Effect of BWR Startup Environments on Crack Growth in Structural Alloys," Trans. Of ASME, Vol. 108, January 1986.

5.2-2 F. P. Ford and M. J. Povich, "The Effect of Oxygen/Temperature Combinations on the Stress Corrosion Susceptibility of Sensitized T-304 Stainless Steel in High Purity Water," Paper 94 presented at Corrosion 79, Atlanta, GA, March 1979.

5.2-3 "BWR Water Chemistry Guidelines - 2004 Revision," EPRI TR-1008192, October 2004.

5.2-4 B. M. Gordon, "The Effect of Chloride and Oxygen on the Stress Corrosion Cracking of Stainless Steels: Review of Literature," Material Performance, NACE, Vol. 19, No. 4, April 1980.

5.2-5 W. J. Shack, et. al., "Environmentally Assisted Cracking in Light Water Reactors: Annual Report, October 1983 – September 1984," NUREG/CR-4287, ANL-85-33, June 1985.

5.2-6 K. S. Brown and G. M. Gordon, "Effects of BWR Coolant Chemistry on the Propensity of IGSCC Initiation and Growth in Creviced Reactor Internal Components," paper presented at the Third International Symposium of Environmental Degradation of Materials in Nuclear Power Systems, ANS-NACE-TMS/AIME, Traverse City, MI, September 1987.

5.2-7 B. M. Gordon et al, "EAC Resistance of BWR Materials in HWC," Proceedings of Second International Symposium Environmental Degradation of Materials in Nuclear Power Systems, ANS, LaGrange Park, IL, 1986.

5.2-8 B. M. Gordon et al, "Hydrogen Water Chemistry for BWRs – Material Behavior," EPRI NP-5080, Palo Alto, CA, March 1987.

Table 5.2-1

Reactor Coolant Pressure Boundary Components (Applicable Code Cases)

Number	Title	Applicable Equipment	Remarks
N-60-5	Material for Core Support Structures, Section III, Division 1	Core Support	Accepted per RG 1.84
N-71-17	Additional Materials for Subsection NF, Classes 1, 2, 3 and MC Component Supports Fabricated by Welding, Section III, Division I.	Component Support	Conditionally Accepted per RG 1.84
N-122-1	Stress Indices for Structure Attachments, Class 1, Section III, Division 1.	Piping	Accepted per RG 1.84
N-247	Certified Design Report Summary for Component Standard Supports, Section III, Division 1, Classes 1, 2, 3 and MC.	Component Support	Accepted per RG 1.84
N-249-14	Additional Material for Subsection NF, Classes 1, 2, 3 and MC Component Supports Fabricated Without Welding, Section III, Division 1.	Component Support	Conditionally Accepted per RG 1.84
N-318-5	Procedure for Evaluation of the Design of Rectangular Cross-Section Attachments on Class 2 or 3 Piping, Section III, Division 1.	Piping	Conditionally Accepted per RG 1.84
N-319-3	Alternate Procedure for Evaluation of Stress in Butt Weld Elbows in Class 1 Piping, Section III, Division 1.	Piping	Accepted per RG 1.84
N-391-2	Procedure for Evaluation of the Design of Hollow Circular Cross-Section Welded Attachments on	Piping	Accepted per RG 1.84

Table 5.2-1

Reactor Coolant Pressure Boundary Components (Applicable Code Cases)

Number	Title	Applicable Equipment	Remarks
	Class 1 Piping. Section III, Division 1.		
N-392-3	Procedure for Evaluation of the Design of Hollow Circular Cross-Section Welded Attachments on Classes 2 and 3 Piping, Section III, Division 1.	Piping	Accepted per RG 1.84
N-580-1	Use of Alloy 600 With Columbium Added, Section III, Division 1.	Core Support	Accepted per RG 1.84
N-608	Applicable Code Edition and Addenda, NCA-1140(a)(2), Section III, Division 1.	All Code Components	Accepted per RG 1.84
N-632	Use of ASTM A 572, Grades 50 and 65 for Structural Attachments to Class CC Containment Liners, Section III, Division 2.	Containment	Accepted per RG 1.84
N-634	Alternatives to the Provisions of CC-2511 for Structural Attachments to Class CC Containment Liners, Section III, Division 2.	Containment	Not Listed in RG 1.84
N-236-1	Repair and Replacement of Class MC Vessels	Containment	Conditionally Accepted Per RG 1.147
N-307-2	Revised Examination Volume for Class 1 Bolting, Table IWB-2500-1, Examination Category B-G-1, when the Examinations are Conducted from the Drilled Hole	RPV Studs	Accepted per RG 1.147

Table 5.2-1

Reactor Coolant Pressure Boundary Components (Applicable Code Cases)

Number	Title	Applicable Equipment	Remarks
N-416-2	Alternative Rules for Hydrostatic Testing of Repair or Replacement of Class 2 Piping	Piping	Conditionally Accepted Per RG 1.147
N-435-1	Alternative Examination Requirements for Vessels with Wall Thicknesses 2 in. or Less	Class 2 Vessels	Accepted Per RG 1.147
N-457	Qualification Specimen Notch Location for Ultrasonic Examination of Bolts and Studs	Bolts and Studs	Accepted Per RG 1.147
N-460	Alternative Examination Coverage for Class 1 and 2 Welds	Class 1 & 2 Components and Piping	Accepted Per RG 1.147
N-463-1	Evaluation Procedures and Acceptance Criteria for Flaws in Class 1 Ferritic Piping that Exceed the Acceptance Standards of IWB-3514-2	Piping	Accepted Per RG 1.147
N-479-1	Boiling Water Reactor (BWR) Main Steam Hydrostatic Test	Main Steam System	Accepted Per RG 1.147
N-491-2	Alternative Rules for Examination of Class 1, 2, 3 and MC Component Supports of Light Water Cooled Power Plants	Component Supports	Not Listed in RG 1.147

Table 5.2-2

Safety/-Relief Valve and Depressurization Valve Settings and/or Capacities

Valve Type SRV /DPV	**Number of Valves**[1]	**Spring Setpoint Maximum Safety Analytical Limit** MPa gauge (psig)	**ASME Rated Capacity at 103% of Safety Analytical Limit Spring Setpoint Pressure** (kg/s each)
ADS SRV	10	8.618 (1250)	124
Non-ADS SRV	8	8.756 (1270)	126
DPV	8	NA	239 [2]

(1) The SRVs also perform the automatic depressurization function.

(2) Minimum capacity in ADS mode.

Table 5.2-3

Systems That May Initiate or Trip During Overpressure Event

Systems	**Initiating/Trip Signal**
Reactor Protection	Reactor shutdown on high flux
IC	Initiated on high reactor pressure or reactor isolation or low reactor water level when mode switch is in “run”
CRD	ON when reactor water level is at L2
RWCU/SDC	OFF when reactor water level is at L2

Table 5.2-4

Reactor Coolant Pressure Boundary Materials

Component	Form	Material	Specification (ASTM/ASME)
Main Steam Isolation Valves (MSIVs)			
Valve Body	Cast	Carbon steel	SA352 LCB
Cover	Forged	Carbon Steel	SA350LF2
Poppet	Forged	Carbon Steel	SA350LF2
Valve stem	Rod	Precipitation-hardened steel	SA 564 Gr 630 (H1100)
Body bolt	Bolting	Alloy steel	SA 540 B23 CL5
Hex nuts	Bolting Nuts	Alloy steel	SA 194 GR7
Safety/-Relief and Depressurization Valves			
Body (SRV)	Forging or Casting	Carbon steel Carbon steel	ASME SA 350 LF2 ASME SA 352 LCB
Body (DPV)	Forging or Casting	Stainless Steel	SA 182, F304L or F316L
Bonnet (yoke)	Forging or Casting	Carbon steel Carbon steel	ASME SA 350 LF2 ASME SA 352 LCB
Nozzle (seat)	Forging or Casting	Stainless steel Carbon steel	ASME SA 182 Gr F316 ASME SA 350 LF2
Body to bonnet stud	Bar/rod	Alloy steel	ASME SA 193 Gr B7
Body to bonnet nut	Bar/rod	Alloy steel	ASME SA 194 Gr 7
Disk	Forging or Casting	Nickel alloy Stainless steel	ASME SA 637 Gr 718 ASME SA 351 CF 3A

Table 5.2-4

Reactor Coolant Pressure Boundary Materials

Component	Form	Material	Specification (ASTM/ASME)
Spring washer and Adjusting Screw or Setpoint adjustment assembly	Forging Forgings	Carbon steel Alloy steel Carbon and alloy steel parts	ASME SA 105 ASME SA 193 Gr B6 Multiple specifications
Spindle (stem)	Bar	Precipitation-hardened steel	ASTM A564 Gr 630 (H1100)
Spring	Wire or Bellville washers	Steel Alloy steel	ASTM A304 Gr 4161 N 45 Cr Mo V67
Main Steam Piping			
Pipe	Seamless	Carbon steel	SA 333 Gr. 6
Contour nozzle	Forging	Low alloy steel	SA 508 Class 3
200 mm 1500 lb. large groove flange	Forging	Carbon steel	SA 350 LF 2
50 mm special nozzle	Forging	Carbon steel	SA 350 LF 2
Elbow	Seamless	Carbon steel	SA 420
Head fitting/penetration piping	Forging	Carbon steel	SA 350 LF 2

CRD			
Middle flange	Forging	Stainless steel	SA-182, Type or SA-336, Class F304/F304L/F316/F316L
Spool piece	Forging	Stainless steel	SA-182, Type or SA-336 Class F304/F304L/F316/F316L

Mounting bolts	Bolting	Alloy steel	SA-193, Grade B7
Reactor Pressure Vessel			
Shells and Heads	Plate	Mn-1/2 Mo-1/2 Ni	SA-533, Type B, Class 1
	Forging	3/4 Ni-1/2 Mo-Cr-V Low alloy steel	SA-508, Grade 3, Class 1
Shell and Head Flange	Forging	3/4Ni-1/2Mo-Cr-V Low alloy steel	SA-508, Grade 3, Class 1
Nozzles	Forging	3/4Ni-1/2Mo-Cr-V Low alloy steel	SA-508, Grade 3, Class 1
Drain Nozzles	Forging	Cr-Ni-Mo Stainless steel	SA-182, Type or SA-336, Class F304/F304L/F316/F316L
Instrumentation Nozzles	Forging	Cr-Ni-Mo Stainless steel and Ni-Cr-Fe	SA-182, Type or SA-336, Class F304/F304L/F316/F316L and Code Case N-580-1
Stub Tubes	Bar, Smls. Pipes Forging	Ni-Cr-Fe	Code Case N-580-1
Isolation Condenser			
Steam pipe	Seamless	Carbon steel	SA-333, Grade 6
Condensate pipe	Seamless	Stainless steel	Type 316L
Feedwater Piping			
Pipe	Seamless	Low Alloy	SA-335, Grade P22
Fittings	Forging	Low Alloy	SA-336, Grade F22

Table 5.2-5

Expected ESBWR Water Chemistry

	Concentration [1] (ppb)					Conductivity
	Iron	**Copper**	**Chloride**	**Sulfate**	**Oxygen** [2]	µS/cm at 25ºC
Condensate	< 20	< 0.5	< 4.0	< 4.0	< 15	~0.075
Condensate Treatment Effluent and Feedwater	< 0.50	< 0.010	< 0.16	< 0.16	30-200 Target < 100	< 0.057
Reactor Water:						
(a) Normal Operation	< 5.0	< 0.50	< 5.0	< 5.0	[2]	< 0.10
(b) Shutdown	< 20	< 1.0	< 5.0	< 5.0	-	< 1.2
(c) Hot Standby	< 5.0	< 0.50	< 5.0	< 5.0	< 300	< 0.10
(d) Depressurized	< 5.0	< 0.50	< 5.0	< 5.0	< 300	< 0.10
Control Rod Drive Cooling Water	< 0.50	< 0.010	< 0.16	< 0.16	30-200 Target < 100	≤ 0.057

Notes:

(1) These limits should be met at least 90% of the time.

(2) Some revision of oxygen values may be established after hydrogen water chemistry has been established.

Table 5.2-6

LD&IS Control and Isolation Functions vs. Monitored Variables

Monitored Variables	LD&IS Isolation Functions									
	Main Steam & Drain Lines	RWCU/ SDC Lines	IC System Lines	Fission Products Sampling Lines	DW LCW Sump Drain Line	DW HCW Sump Drain Line	Contain-ment Purge & Vent Valves	RCCWS Lines to DW Air Coolers	FAPCS Process Lines	R/B HVAC Exhaust Ducts
RWCU/SDC Flow High		X								
HCW Drain Line Radiation High						X				
LCW Drain Line Radiation High					X					
SLC Initiation Signal		X								
Refueling Area Air Exhaust Radiation High							X			X
Reactor Building Air Exhaust Radiation High							X			X
IC Condensate Flow High		X								
IC Steam Flow High		X								
Drywell Pressure High				X	X	X	X	X	X	X

Table 5.2-6

LD&IS Control and Isolation Functions vs. Monitored Variables

Monitored Variables	**LD&IS Isolation Functions**									
	Main Steam & Drain Lines	RWCU/ SDC Lines	IC System Lines	Fission Products Sampling Lines	DW LCW Sump Drain Line	DW HCW Sump Drain Line	Contain-ment Purge & Vent Valves	RCCWS Lines to DW Air Coolers	FAPCS Process Lines	R/B HVAC Exhaust Ducts
Main Condenser Vacuum Low	X									
Turbine Area Ambient Temperature High	X									
MSL Tunnel Ambient Temperature High	X	X								
IC Pool Vent Radiation High			X							
MSL Flow Rate High	X									
Turbine Inlet Pressure Low	X									
Reactor Water Level Low (L1, L2)	X	X		X	X	X	X	X	X	X

Table 5.2-7

Leakage Sources vs. Monitored Variables

Monitored Variables (2)	**Leakage Source**																								
	Loca-tion (1)	Main Steam-lines		IC Steam-lines		IC Conden-sate Lines		RCCWS Lines		FAPCS Lines		RWCU/ SDC Lines		Feed-water Lines		GDCS Water		Reactor Vessel Head Seal		Valve Stem Packing		Misc. Leaks	
		I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O
Inter-System Leakage Radiation High													X												
RWCU/SDC Flow High													X												
Equip. Areas Differential Temperature High			X										X												
MSL Tunnel or Turbine Building Area Ambient Temperature High			X										X		X										
MSL or IC Steamline Flow High		X	X	X	X																				
Drywell Air Cooler Cond. Flow High		X		X				X		X		X		X		X									
Vessel Head Flange Seal Pressure High																		X							

Table 5.2-7

Leakage Sources vs. Monitored Variables

Monitored Variables (2)	**Leakage Source**																						
	Loca-tion (1)	Main Steam-lines		IC Steam-lines		IC Conden-sate Lines		RCCWS Lines		FAPCS Lines		RWCU/ SDC Lines		Feed-water Lines		GDCS Water		Reactor Vessel Head Seal		Valve Stem Packing		Misc. Leaks	
		I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O
Reactor Building Equip./Floor Drain Sump Pump Activity			X		X		X		X		X		X		X						X		X
SRV Discharge Line Temperature High		X																					
Drywell Temperature High		X		X				X		X		X		X		X						X	
Drywell Fission Product Radiation High		X		X		X						X		X									
Drywell Equip. Drain Sump Level Change High																		X		X			
Drywell Floor Drain Sump Level Change High		X		X		X		X		X		X		X		X						X	
Drywell Pressure High		X		X								X		X									
Reactor Water Level Low (L1, L2)		X	X	X	X	X	X	X		X		X	X	X									

(1) I = Inside Drywell Leakage; O = Outside Drywell Leakage

(2) X = Alarm is provided for this monitored variable.

Table 5.2-8

LD&IS Control and Isolation Functions vs, Monitored Variables

Monitored Variables	LD&IS Isolation Functions									
	Main Steam & Drain Lines	RWCU/ SDC Lines	IC System Lines	Fission Products Sampling Lines	DW LCW Sump Drain Line	DW HCW Sump Drain Line	Contain-ment Purge & Vent Valves	RCCWS Lines to DW Air Coolers	FAPCS Process Lines	R/B HVAC Exhaust Ducts
RWCU/SDC Flow High		X								
HCW Drain Line Radiation High						X				
LCW Drain Line Radiation High					X					
SLC Initiation Signal		X								
Refueling Area Air Exhaust Radiation High							X			X
Reactor Building Air Exhaust Radiation High							X			X
IC Condensate Flow High		X								
IC Steam Flow High		X								
Drywell Pressure High				X	X	X	X	X	X	X

Monitored Variables	LD&IS Isolation Functions									
	Main Steam & Drain Lines	RWCU/ SDC Lines	IC System Lines	Fission Products Sampling Lines	DW LCW Sump Drain Line	DW HCW Sump Drain Line	Contain- ment Purge & Vent Valves	RCCWS Lines to DW Air Coolers	FAPCS Process Lines	R/B HVAC Exhaust Ducts
Main Condenser Vacuum Low	X									
Turbine Area Ambient Temperature High	X									
MSL Tunnel Ambient Temperature High	X	X								
IC Pool Vent Radiation High			X							
MSL Flow Rate High	X									
Turbine Inlet Pressure Low	X									
Reactor Water Level Low (L1, L2)	X	X		X	X	X	X	X	X	X

Table 5.2-9

Leakage Sources vs. Monitored Variables

Monitored Variables (2)	**Leakage Source**																								
	Loca-tion (1)	Main Steam-lines		IC Steam-lines		IC Conden-sate Lines		RCCWS Lines		FAPCS Lines		RWCU/ SDC Lines		Feed-water Lines		GDCS Water		Reactor Vessel Head Seal		Valve Stem Packing		Misc. Leaks	
		I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O
Inter-System Leakage Radiation High													X										
RWCU/SDC Flow High													X										
Equip. Areas Differential Temperature High			X										X										
MSL Tunnel or Turbine Building Area Ambient Temperature High			X										X		X								
MSL or IC Steamline Flow High		X	X	X	X																		
Drywell Air Cooler Cond. Flow High		X		X				X		X		X		X		X							
Vessel Head Flange Seal Pressure High																		X					

Table 5.2-9

Leakage Sources vs. Monitored Variables

Monitored Variables (2)	Loca-tion (1)	**Leakage Source** Main Steam-lines		IC Steam-lines		IC Conden-sate Lines		RCCWS Lines		FAPCS Lines		RWCU/ SDC Lines		Feed-water Lines		GDCS Water		Reactor Vessel Head Seal		Valve Stem Packing		Misc. Leaks	
		I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O
Reactor Building Equip./Floor Drain Sump Pump Activity			X		X		X		X		X		X		X						X		X
SRV Discharge Line Temperature High		X																					
Drywell Temperature High		X		X				X		X		X		X		X						X	
Drywell Fission Product Radiation High		X		X		X						X		X									
Drywell Equip. Drain Sump Level Change High																		X		X			
Drywell Floor Drain Sump Level Change High		X		X		X		X		X		X		X		X						X	
Drywell Pressure High		X		X								X		X									
Reactor Water Level Low (L1, L2)		X	X	X	X	X	X	X		X		X	X	X									

(1) I = Inside Drywell Leakage; O = Outside Drywell Leakage

(2) X = Alarm is provided for this monitored variable.

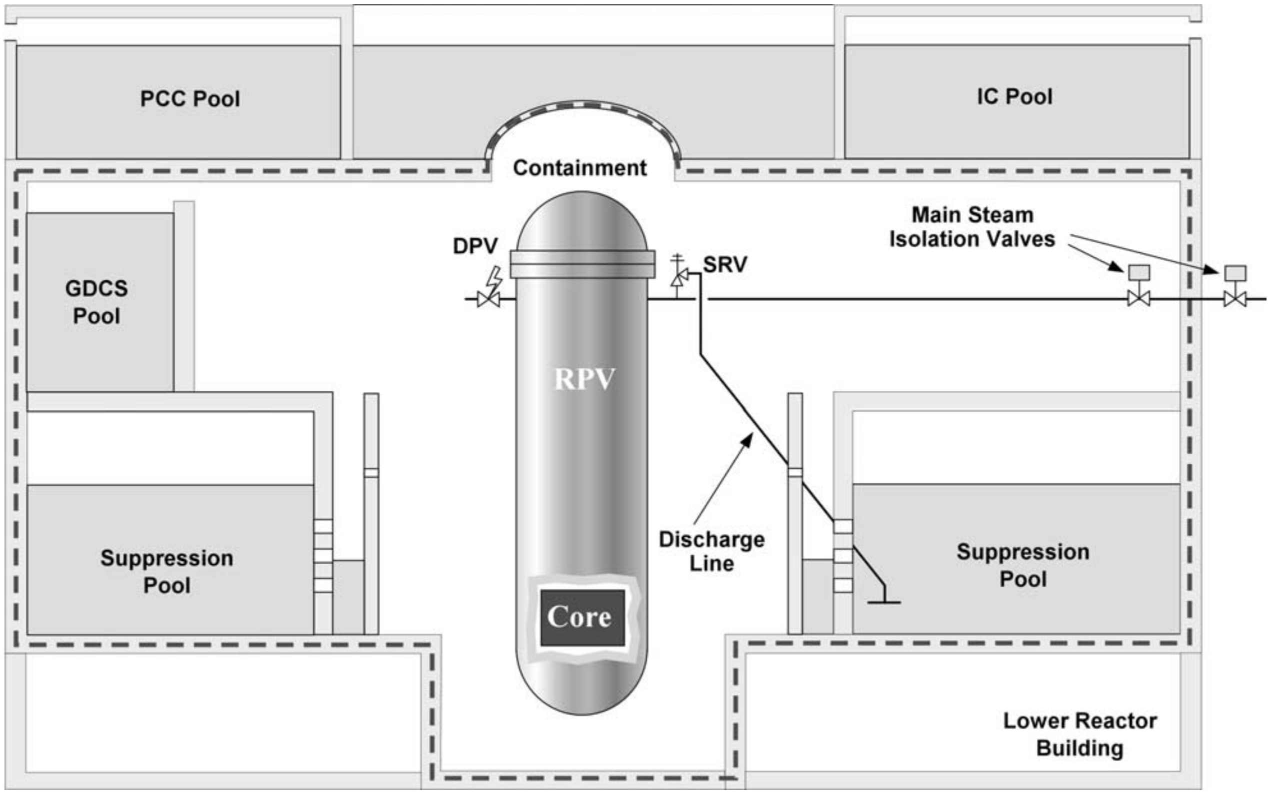


Figure 5.2-1. Safety/Relief Valve Schematic Elevation

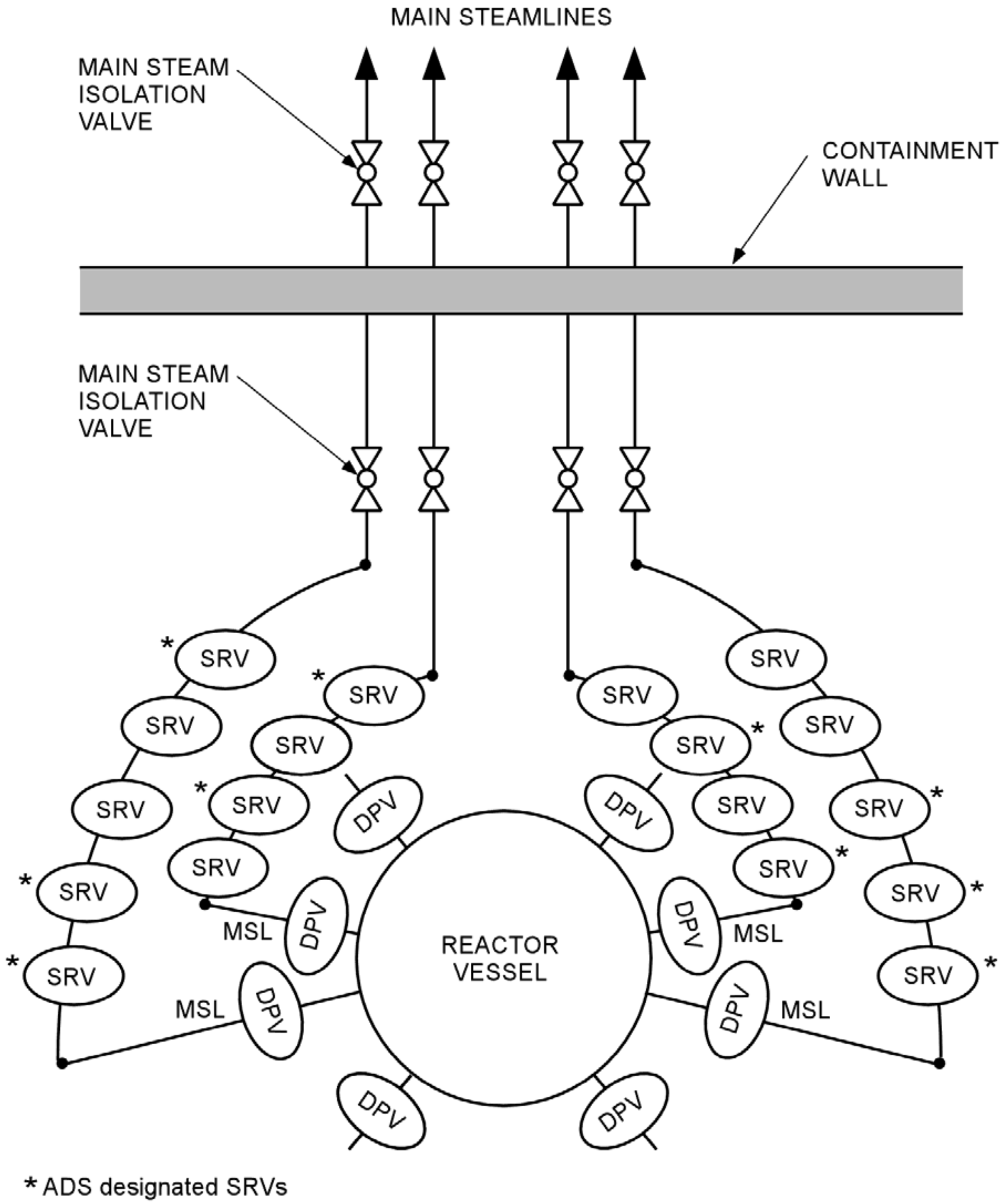


Figure 5.2-2. Safety-Relief Valves, Depressurization Valves and Steamline Diagram

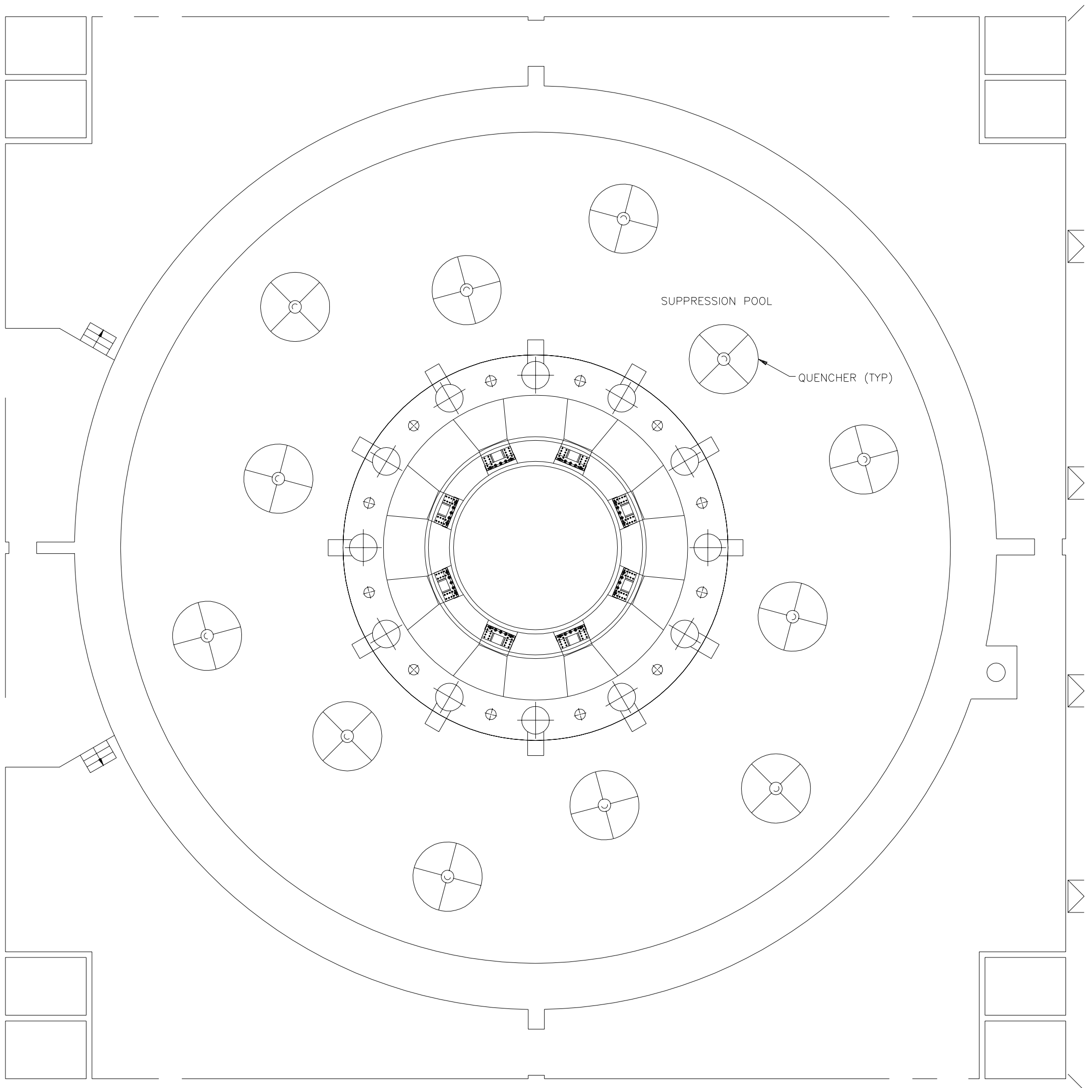


Figure 5.2-3. Safety-Relief Valve Discharge Line Quencher Arrangement

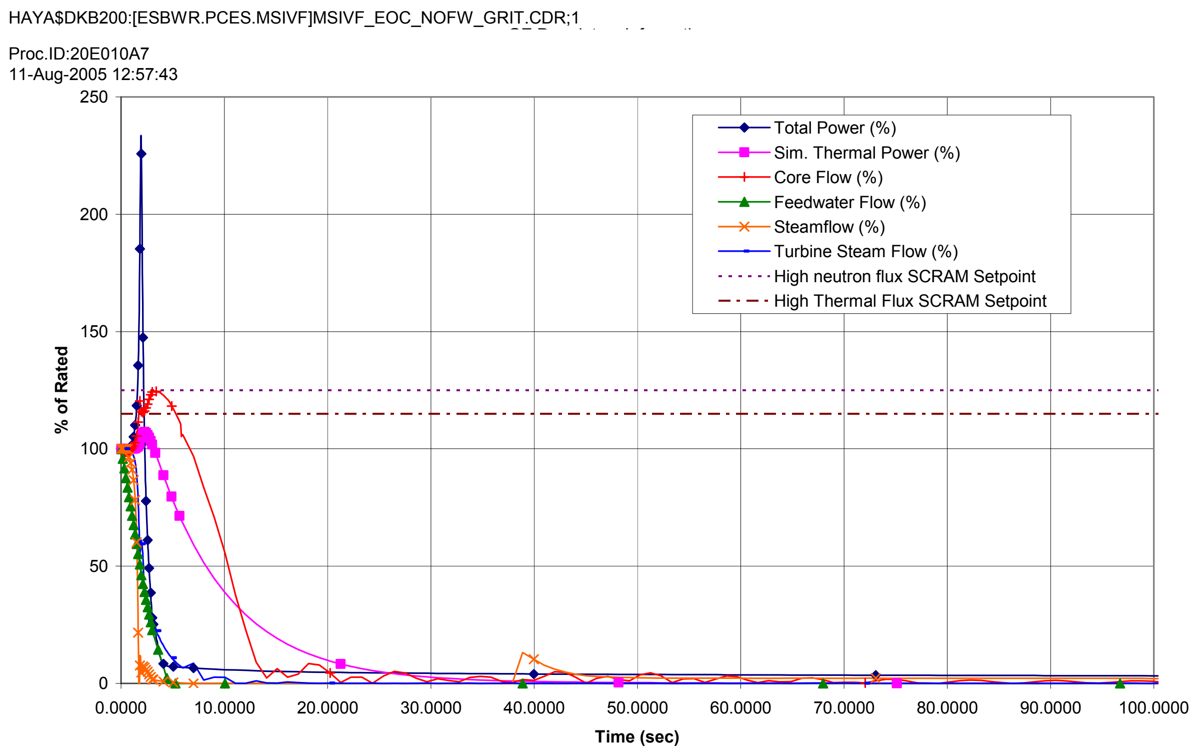


Figure 5.2-4a. MSIV Closure – Flux Scram

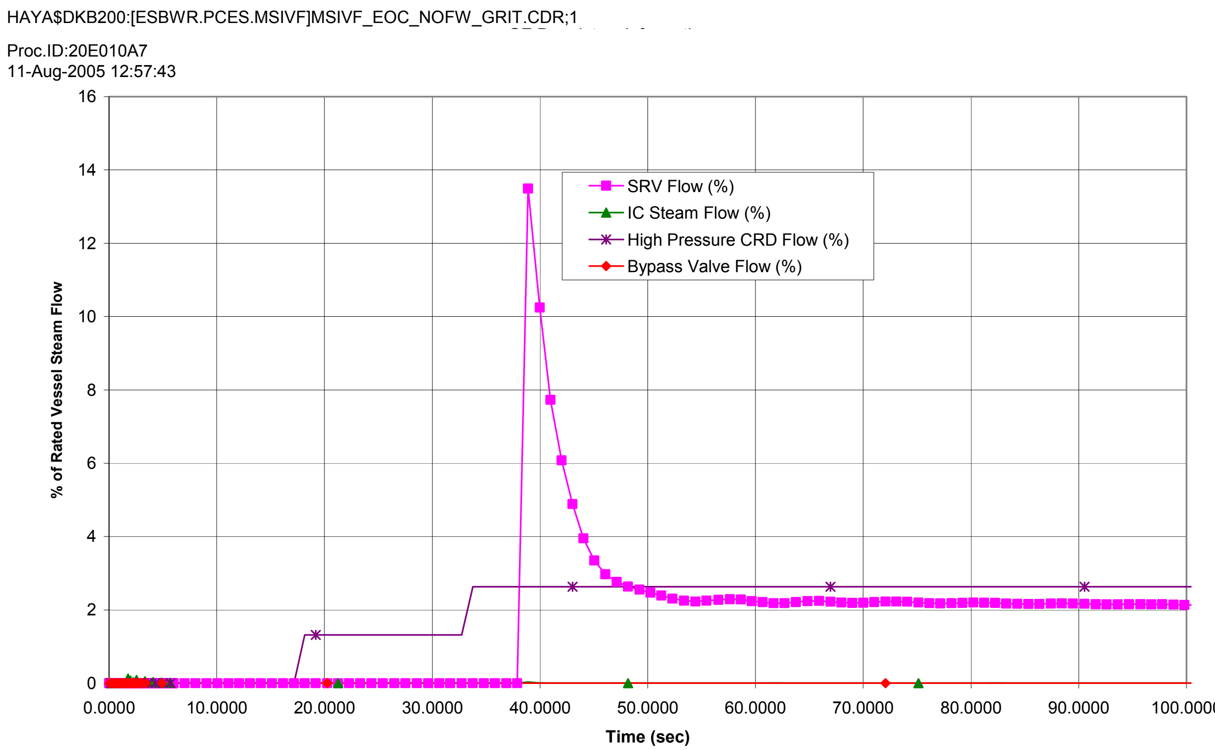


Figure 5.2-4b. MSIV Closure – Flux Scram

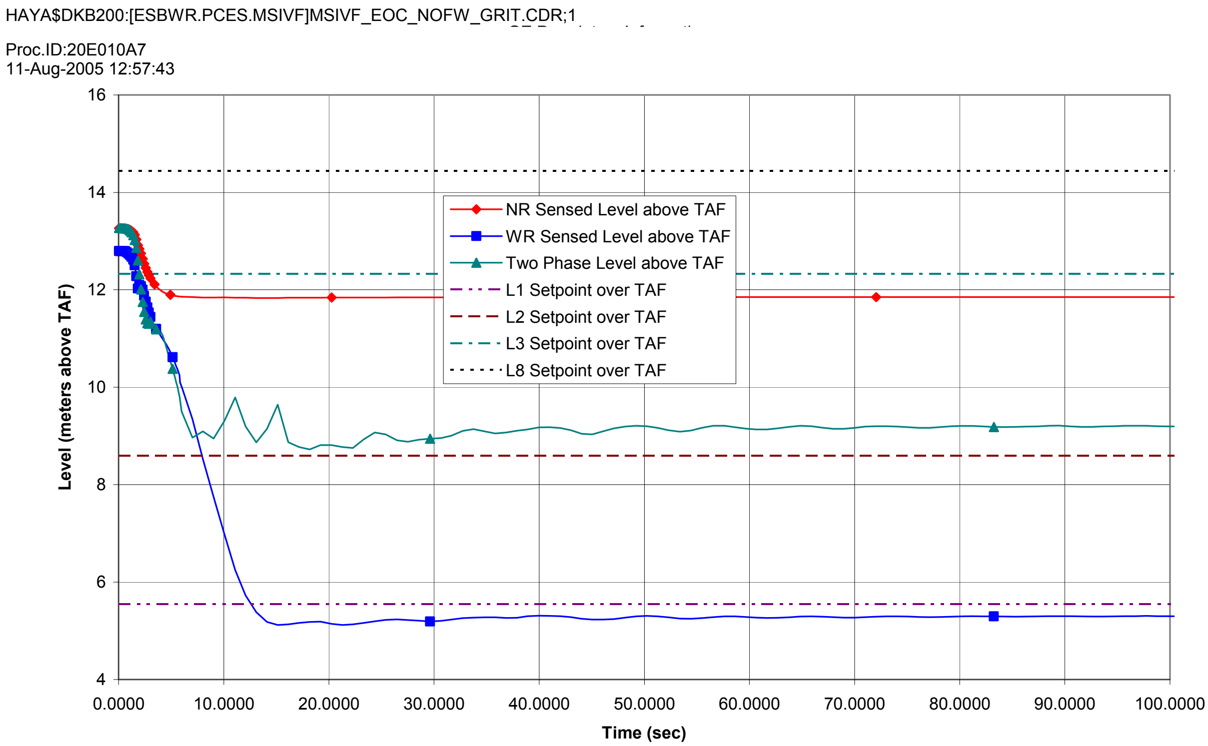


Figure 5.2-4c. MSIV Closure – Flux Scram

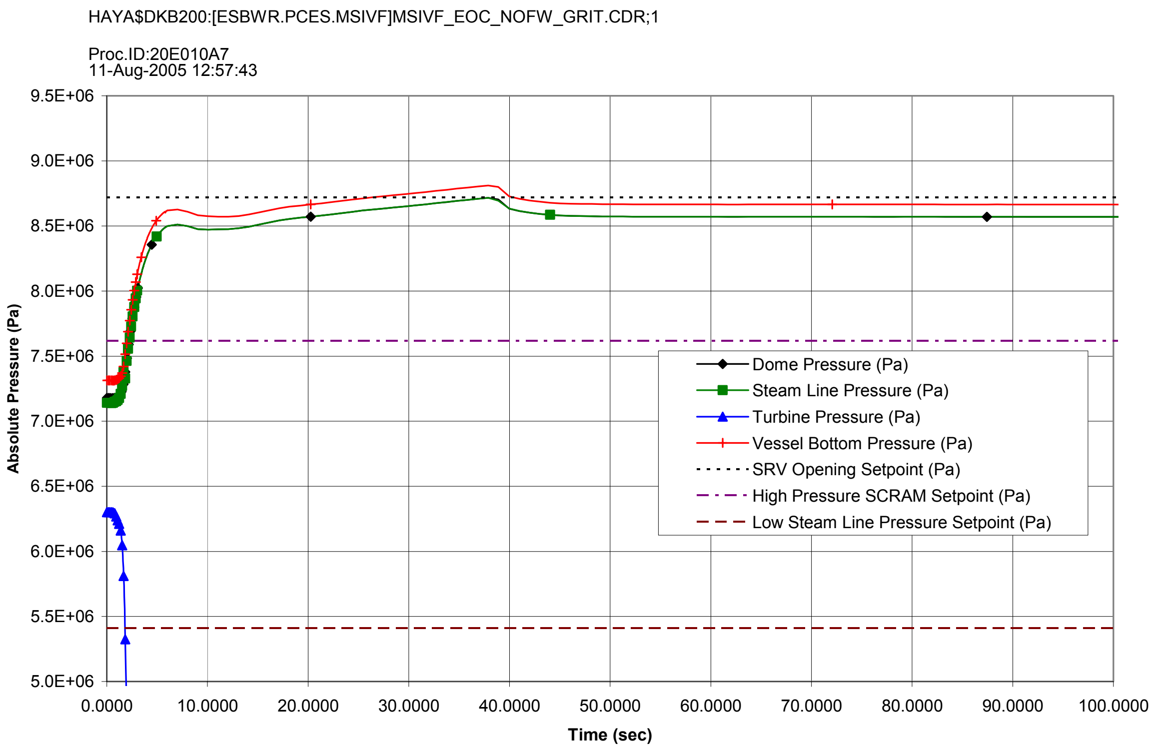


Figure 5.2-4d. MSIV Closure – Flux Scram

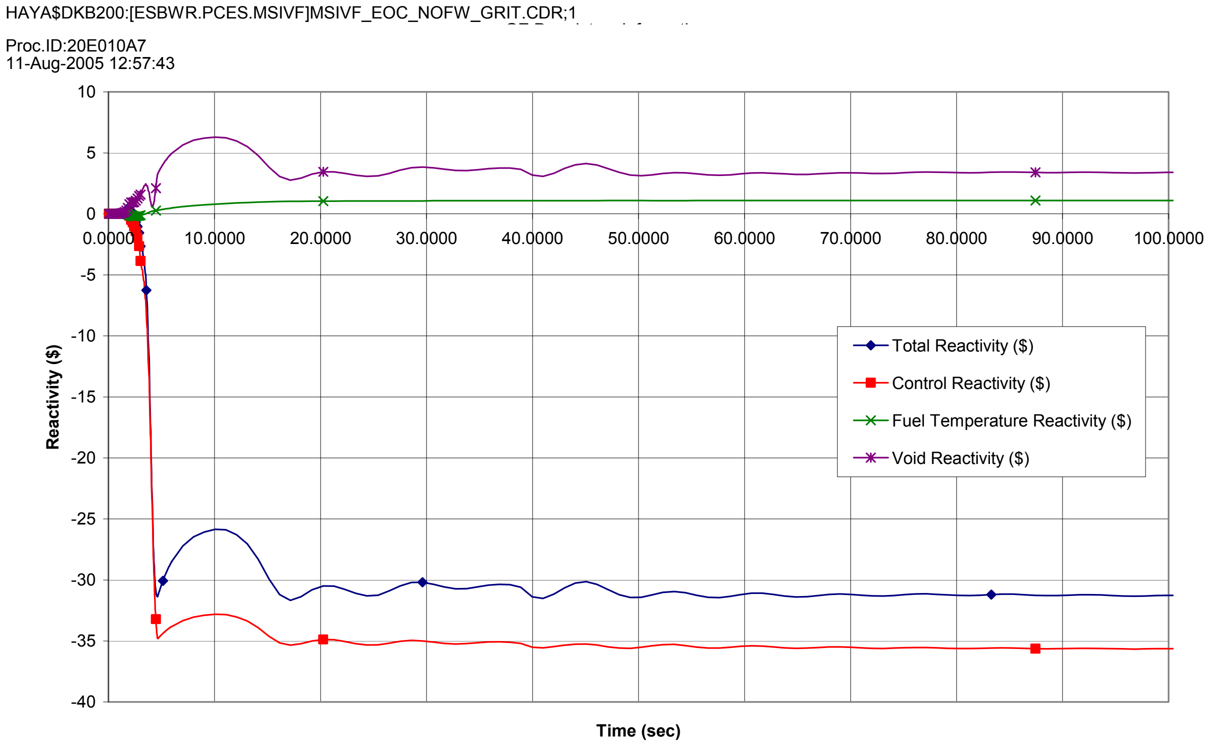


Figure 5.2-4e. MSIV Closure – Flux Scram

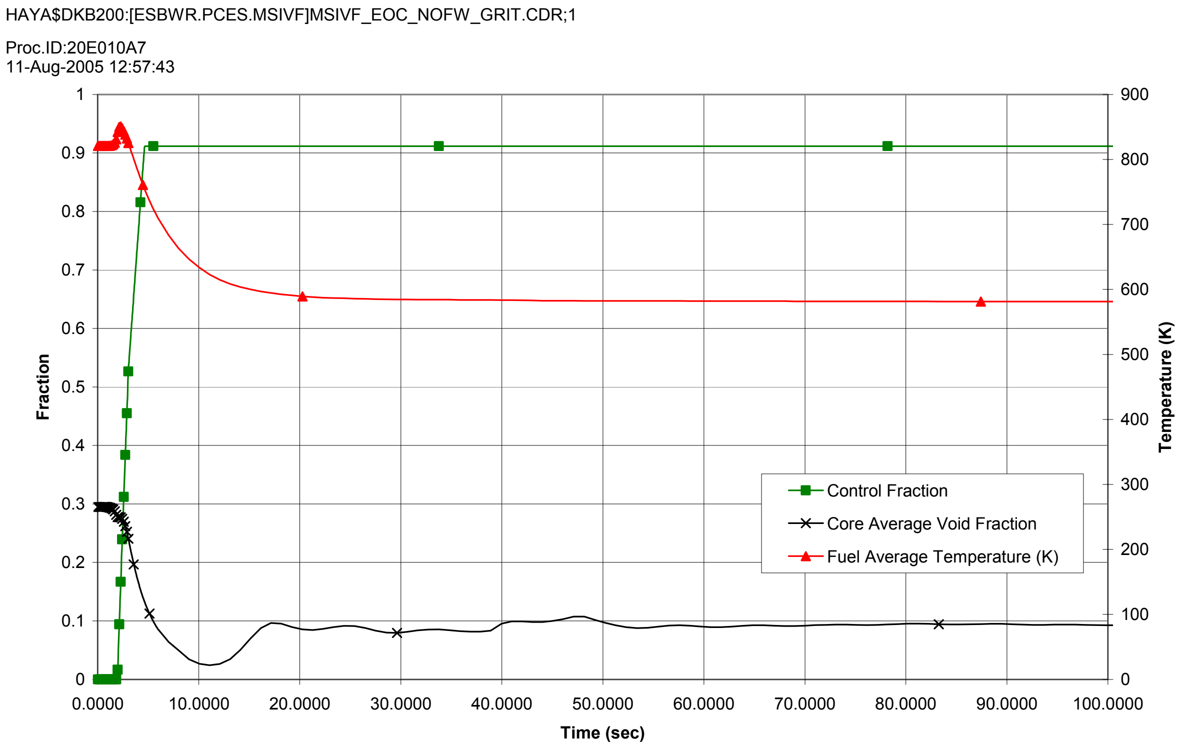


Figure 5.2-4f. MSIV Closure – Flux Scram

5.3 REACTOR VESSEL

5.3.1 Reactor Vessel Materials

This Subsection is written in the format and meets the requirements of SRP 5.3.1 Draft Rev. 2. The ESBWR meets the requirements of:

- GDC 1 and 30, as they relate to quality standards for design, fabrication, erection, and testing of structures, systems and components;
- GDC 4, as it relates to compatibility of components with environmental conditions;
- GDC 14, as it relates to prevention of rapidly propagating fractures of the RCPB;
- GDC 31, as it relates to material fracture toughness;
- GDC 32, as it relates to the requirements for a materials surveillance program;
- 10 CFR 50.55a as it relates to quality standards for design and determination and monitoring of fracture toughness;
- 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation", as it relates to reactor coolant pressure boundary fracture toughness and material surveillance requirements of 10 CFR 50, Appendix G and Appendix H;
- 10 CFR 50, Appendix B, Criterion XIII, as it relates to onsite material cleaning control;
- 10 CFR 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness; and
- 10 CFR 50, Appendix H, as it relates to the determination and monitoring of fracture toughness.

The specific criteria which meet the relevant requirements are as presented in the following subsections.

5.3.1.1 Materials Specifications

The materials used in the reactor pressure vessel (RPV) and appurtenances are shown in Table 5.2-4, together with the applicable specifications.

The RPV materials shall comply with the provisions of ASME Section III, and shall also meet the requirements of ASME Code Section II materials and 10 CFR 50, Appendix G. The RPV materials also meet the additional requirements as explained in the following subsections.

These materials provide adequate strength, fracture toughness, fabricability, and compatibility with the BWR environment. Their suitability has been demonstrated by long-term successful operating experience in reactor service.

5.3.1.2 Special Procedures Used for Manufacturing and Fabrication

The RPV is constructed primarily from low alloy, high strength steel plate and forgings. Plates are ordered to ASME SA-533, Type B, Class 1, and forgings to ASME SA-508, Grade 3, Class 1. These materials are melted to fine grain practice and are supplied in the quenched and tempered condition. Further restrictions include a requirement for vacuum degassing to lower the hydrogen level and improve the cleanliness of the low alloy steels. Specified limits for materials used in the core beltline region are presented in Table 5.3-1.

Studs, nuts, and washers for the main closure flange have special material controls as presented in Table 5.3-1. Welding electrodes for low alloy steel are low hydrogen type ordered to ASME SFA-5.5, and weld filler metal to SFA-5.23 and SFA-5.28.

All plate, forgings, and bolting are 100% ultrasonically tested and surface examined by magnetic particle methods or liquid penetrant methods as required by ASME Code Section III, Division 1.

Fracture toughness properties of materials are also measured and controlled in accordance with ASME Code Section III, Division 1.

All fabrication of the RPV is performed in accordance with GE approved drawings, fabrication procedures, and test procedures. The shells and vessel heads are made from formed plates or forgings, whereas flanges and nozzles are made from forgings. Welding performed to join these vessel components is in accordance with procedures qualified per ASME Code Section III and IX requirements. Weld test samples are required for each procedure used on major vessel full penetration welds. Tensile and impact tests are performed to determine the properties of the base metal, heat-affected zone (HAZ), and weld metal.

Gas Tungsten Arc Welding (GTAW), Gas Metal Arc Welding (GMAW), Shielded Metal Arc Welding (SMAW), and Submerged Arc Welding (SAW) processes may be employed. Electroslag welding is not used except for cladding. Preheat and interpass temperatures employed for welding of low alloy steel meet or exceed the values given in ASME Code Section III, Appendix D. Post-weld heat treatment of all low alloy welds is performed in accordance with ASME Code, Subsection NB-4620 (see Table 5.3-1).

Volumetric examination and surface examination are performed on all pressure-retaining welds as required by ASME Code Section III, Subsection NB-5320. In addition, all pressure-retaining welds are given a supplemental ultrasonic pre-service examination in accordance with ASME Section XI.

The materials, fabrication procedures, and testing methods used in the construction of the ESBWR reactor pressure vessel meet or exceed requirements of ASME Code Section III, Class 1 vessels.

5.3.1.3 Special Methods for Nondestructive Examination

The materials and welds on the RPV are examined in accordance with methods prescribed and meet the acceptance requirements specified by ASME Section III. The pressure-retaining welds are volumetrically examined. In addition, the pressure-retaining welds are ultrasonically examined using acceptance standards that are equivalent or more restrictive than required by ASME Code Section XI. The ultrasonic examination method, including calibration, instrumentation, scanning sensitivity, and coverage, is based on the requirements imposed by ASME Code Section XI, Appendix I.

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

Regulatory Guide 1.31: Control of Stainless Steel Welding

Controls on stainless steel welding are discussed in Subsection 5.2.3.4.2. Consistent with Generic Letter 88-01 and NUREG-0313 Revision 2, control of weld filler metal ferrite content is described in Paragraph 5.2.3.4.1.

Regulatory Guide 1.34: Control of Electroslag Weld Properties

The requirements of this regulatory guide are not applicable to the ESBWR vessel, because electroslag welding is not employed in structural welds of low alloy steel. Electroslag welding is not used except for cladding.

Regulatory Guide 1.43: Control of Stainless Steel Weld Cladding of Low Alloy Steel Components

Regulatory Guide 1.43 is concerned with cracking of low alloy steels underneath stainless steel weld deposited cladding. The requirements of this Regulatory Guide are not applicable to the ESBWR vessel because the RPV is constructed from low alloy steel forgings or plates conforming to SA-508, Grade 3 or SA-533, Type B, which are produced to fine grain practice. Therefore, underclad cracking is not a concern, and the requirements of this regulatory guide are not applicable.

Regulatory Guide 1.44: Control of the Use of Sensitized Stainless Steel

Sensitization of stainless steel is controlled by the use of service proven low carbon materials and by use of appropriate design and processing steps, including solution heat treatment, control of welding heat input, control of heat treatment during fabrication and control of stresses. As more completely described in Paragraph 5.2.3.4.1, these controls conform to the guidance of Generic Letter 88-01 and NUREG-0313 Revision 2.

Regulatory Guide 1.50: Control of Preheat Temperature For Welding Low Alloy Steel

Regulatory Guide 1.50 delineates preheat temperature control requirements and welding procedure qualifications supplementing those in ASME Code Sections III and IX. Except as noted below, Regulatory Guide 1.50 is followed.

Preheat temperature employed for welding of low alloy steel meets or exceeds the recommendations of ASME Code Section III, Appendix D. Components are either held for an extended time at preheat temperature to assure removal of hydrogen, or preheat is maintained until post-weld heat treatment. The minimum preheat and maximum interpass temperatures are specified and monitored.

Acceptance Criterion II.3.b(1)(a) of SRP Subsection 5.2.3 for control of preheat temperature requires that minimum and maximum interpass temperatures be specified. The minimum preheat and maximum interpass temperatures for welding the ESBWR reactor vessel are specified. In addition, welding procedure qualification shall be performed at a temperature within the range of minimum preheat temperature and minimum preheat temperature plus 28°C.

All pressure-retaining welds are nondestructively examined by volumetric methods.

Regulatory Guide 1.71: Welder Qualification for Areas of Limited Accessibility

Qualification for areas of limited accessibility is discussed under Regulatory Guide 1.71 in Subsection 5.2.3.4.2 of this report.

Regulatory Guide 1.99: Effects of Residual Elements on Predicted Radiation Damage to Reactor Pressure Vessel Materials

Predictions for changes in transition temperature and upper shelf energy are made in accordance with the requirements of Regulatory Guide 1.99.

Regulatory Guide 1.37: Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants

The cleaning of systems and components on the site during and at the completion of construction is accomplished to written procedures, which ensures both cleanliness and that the components are not exposed to materials or practices which may degrade their performance. For components containing stainless steel, the procedures shall comply with Regulatory Guide 1.37. The procedures prohibit contact with low melting point compounds, substances which are known to cause stress corrosion cracking or which can release, in any manner, substances that can cause such problems. In addition, there are controls placed on the use of grinding wheels and wire brushes, which assures that they cannot introduce degrading materials either through prior usage or through their materials of construction. In this context, degradation includes stress corrosion cracking. Controls also control introduction of unnecessary dirt and require control of dirt producing processes such as welding or grinding including prompt cleaning.

5.3.1.5 Fracture Toughness

Compliance with 10 CFR 50, Appendix G

Appendix G of 10 CFR 50 is interpreted for Class 1 primary coolant pressure boundary components of the ESBWR design and complied with as discussed in Methods of Compliance below and Subsection 5.3.2. The specific temperature/pressure limits for the operation of the reactor (Figures 5.3-1 and 5.3-2) are based on 10 CFR 50 Appendix G, Paragraph IV, A.2. See Subsection 5.3.4 for COL information requirements concerning fracture toughness data.

Methods of Compliance

The following items are the interpretations and methods used to comply with 10 CFR 50, Appendix G:

- Material Test Coupons and Test Specimens (Appendix G III-A)

  Test coupons are removed from the location in each product form as specified in Subsection NB-2220 of ASME Section III. The heat treatment of the test coupons is performed in accordance with Subarticle NB-2210.

  It is understood that separately produced test coupons in accordance with Subparagraph NB-2223.3 may be used for forgings.

- Location and Orientation of Test Specimens (Appendix G III-A)

  The test specimens are located and oriented in accordance with ASME Code Section III, Subsection NB-2322. Charpy-V impact specimens for testing of plate will be oriented normal to the principal rolling direction (not in the thickness direction). For forged material other than bolting and bars the Charpy-V impact specimens will be oriented normal to the principal direction in which the material was worked. Axial specimens are used for bolting and bars.

In regard to 10 CFR 50 Appendix H, the surveillance test material is selected on the basis of the requirements of ASTM E185-82 and Regulatory Guide 1.99 to provide a conservative adjusted reference temperature for the beltline materials. The weld test plate for the surveillance program specimens has the principal working direction parallel to the weld seam to assure that HAZ specimens are normal to the principal working direction. See Subsection 5.3.4 for COL license information concerning materials and surveillance capsule requirements.

- Records and Procedures for Impact Testing (Appendix G III-C)

  Preparation of impact testing procedures, calibration of test equipment, and retention of the records of these functions and test data comply with the requirements of ASME Code Section III. Personnel conducting impact testing are qualified by experience, training or qualification testing that demonstrates competence to perform tests in accordance with the testing procedure.

- Charpy-V Curves for the RPV Beltline (Appendix G III-A and G IV-A.1)

  A full transverse Charpy-V curve is determined for all heats of base material and weld metal used in the core beltline region with a minimum of three (3) specimens tested in the upper shelf region (>95% shear). The minimum initial upper shelf energy level for base material and weld metal in the beltline region meets or exceeds 102 J (75 ft-lb), and is predicted to maintain a Charpy upper shelf energy of greater than 68 J (50 ft-lb) throughout the life of the vessel, as required by 10 CFR 50, Appendix G-IV-A.1.

  In regard to 10 CFR 50, Appendix G III-A, it is understood that separate, unirradiated baseline specimens per ASTM E-185, Paragraph 6.3.1, are used to determine the transition temperature curve of the core beltline base material, HAZ and weld metal.

- Bolting Material

  All bolting material exceeding 25.4 mm (one-inch) diameter has special material requirements as presented in Table 5.3-1.

- Fracture Toughness Margins in the Control of Reactivity (Appendix G IV-A).

  ASME Code, Section III, Appendix G, was used in conjunction with the defect size of Subsection 5.3.1.5 (6) in determining pressure/temperature limitations for all phases of normal plant operation including anticipated operational occurrences.

5.3.1.6 Material Surveillance

5.3.1.6.1 Compliance with Reactor Vessel Material Surveillance Program Requirements

The materials surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and thermal environment.

Reactor vessel materials surveillance specimens are provided in accordance with requirements of ASTM E 185 and 10 CFR 50 Appendix H. Materials for the program are selected to represent materials used in the reactor beltline region. Specimens are manufactured from a forging actually used in the beltline region and a weld typical of those in the beltline region and thus represent base metal, weld material, and the weld HAZ material. The base metal and weld are heat treated in a manner, which simulates the actual heat treatment performed on the beltline region of the completed vessel. Each in-reactor surveillance capsule contains 36 Charpy V-notch and 6 tensile specimens. The capsule loading consists of 12 Charpy V Specimens each of base metal, weld metal, HAZ material, and three tensile specimens each from base metal and weld metal. A set of out-of-reactor beltline Charpy V-notch specimens, tensile specimens, and archive material are provided with the surveillance test specimens. Neutron dosimeters and temperature monitors are located within the capsules as required by ASTM E 185.

Four capsules are provided to consider the 60 year design life of the vessel. This exceeds the three capsules specified in ASTM E 185 as required by 10 CFR 50, Appendix H, since the predicted transition temperature shift is less than 55.6°C (100°F) at the inside of the vessel.

The following proposed withdrawal schedule is modified from the ASTM E 185 schedule to consider the 60 year design life:

- first capsule: after 6 effective full power years;
- second capsule: after 20 effective full power years;
- third capsule: with an exposure not to exceed the peak EOL fluence;
- fourth capsule: schedule determined based on results of first three capsules per ASTM E 185, Paragraph 7.6.2.

Fracture toughness testing of irradiated capsule specimens are in accordance with requirements of ASTM E 185 as required by 10 CFR 50 Appendix H.

5.3.1.6.2 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in Subsection 4.1.4.5.

5.3.1.6.3 Predicted Irradiation Effects on Beltline Materials

Transition temperature changes and changes in upper shelf energy (USE) are calculated in accordance with the rules of Regulatory Guide 1.99. Reference temperatures are established in accordance with 10 CFR 50 Appendix G, and Subsection NB-2330 of the ASME Code.

Because weld material chemistry and fracture toughness data are not available at this time, the limits in the design document were used to estimate worst case irradiation effects.

These estimates for the adjusted reference temperature and upper shelf energy at end of life for the beltline weld and forging are provided in Table 5.3-2.

5.3.1.6.4 Positioning of Surveillance Capsules and Methods of Attachment (Appendix H.II B (2))

Surveillance specimen capsules are located at four azimuths at a common elevation in the core beltline region. A minimum capsule lead factor of 1 is used in determining the locations of the capsules. The sealed capsules are not attached to the vessel but are in welded capsule holders. The capsule holders are mechanically retained by capsule holder brackets welded to the vessel cladding. Since reactor vessel specifications require that all low alloy steel pressure vessel boundary materials be produced to fine grain practice, underclad cracking is of no concern. The capsule holder brackets allow the removal and reinsertion of capsule holders. Although not

Code parts, these brackets are designed, fabricated, and analyzed to the requirements of ASME Section III. A positive spring-loaded locking device is provided to retain the capsules in position throughout any anticipated event during the lifetime of the vessel. (See Subsection 5.3.4 for COL license information requirements pertaining to materials and surveillance capsules.)

In areas where brackets (such as the surveillance specimen holder brackets) are located, additional nondestructive examinations are performed on the vessel base metal and stainless steel weld deposited cladding or weld buildup pads during vessel manufacture. The base metal is ultrasonically examined by straight beam techniques to a depth at least equal to the thickness of the bracket being joined. The area examined is the area of width equal to at least half the thickness of the part joined. The required stainless steel weld deposited cladding is similarly examined. The full penetration welds are liquid penetrant examined. The minimum cladding thickness is shown in Table 5.3-3. These requirements have been successfully applied to a variety of bracket designs, which are attached to weld deposited stainless steel cladding or weld buildups in many operating BWRs.

5.3.1.6.5 Time and Number of Dosimetry Measurements

GE provides a separate neutron dosimeter so that fluence measurements may be made at the vessel ID during the first fuel cycle to verify the predicted fluence at an early date in plant operation. This measurement is made over this short period to avoid saturation of the dosimeters now available. Once the fluence to thermal power output is verified, no further dosimetry is considered necessary because of the linear relationship between fluence and power output. It is possible, however, to install a new dosimeter, if required, during succeeding fuel cycles.

5.3.1.6.6 Additional Measurement of Fracture Toughness

In addition to the compliance with 10 CFR 50 Appendix G for fracture toughness and 10 CFR 50 Appendix H for surveillance test material of the beltline forging which may be exposed to end-of-life fluence in excess of $1*10^{18}$ nvt, additional surveillance specimens shall be included for determination of upper shelf K_{IC} at RT_{NDT} +33°C (60°F). The J_{IC} method as specified in ASTM E 813 shall be used to determine K_{IC}.

Reactor Vessel Fasteners

The reactor vessel closure head (flange) is fastenened to the reactor vessel shell flange by multiple sets of threaded studs and nuts. The lower end of each stud is installed in a threaded hole in the vessel flange. A nut and washer are installed on the upper end of each stud. The proper amount of preload can be applied to the studs by sequential tensioning using hydraulic tensioners.

Hardness tests are performed on all main closure bolting to demonstrate that heat treatment has been properly performed.

5.3.1.7 Regulatory Guide 1.65

Regulatory Guide 1.65 defines acceptable materials and testing procedures with regard to reactor vessel stud bolting for light-water-cooled reactors.

The design and analysis of reactor vessel bolting materials are in full compliance with ASME Code, Section III, Class I, requirements. The RPV closure studs are SA-540 Grade B23 or B24

(AISI 4340). The maximum allowable ultimate tensile strength is 1172 MPa. Also, the Charpy impact test requirements of NB-2333 are satisfied (the lowest C_V energy is greater than the requirement of 61 J (45 ft-lbf) at the stud preload temperature; the lowest reported C_V expansion exceeds the 0.64 mm (0.025 in.) required).

In regards to regulatory position C.2.b, the bolting materials are ultrasonically examined in accordance with ASME Code, Section III, Subsection NB-2580, after final heat treatment and prior to threading as specified. The examination is in accordance with the requirements of ASME Code, Section II, ASME SA-388. The procedures approved for use in practice are judged to insure comparable material quality and are considered adequate on the basis of compliance with the applicable requirements of ASME Code Subsection NB-2580.

The straight beam examination is performed on 100% of cylindrical surfaces and from both ends of each stud using a 19 mm (0.75 in.) maximum diameter transducer. The reference standard for the radial scan contains a 12.7 mm (0.5 in.) diameter flat bottom hole with a depth of 10% of the thickness. The end scan standard is per ASME SA-388. Surface examinations are performed on the studs and nuts after final heat treatment and threading as specified in the guide, in accordance with ASME SA-388. Any indication greater than that from the applicable calibration feature is unacceptable. The distance/amplitude correction curve for the straight beam end scan of RPV head studs, nuts, and washers is established as follows:

- For cylinders having a length (L) to O.D. ratio of 7 or less, the distance/amplitude curve is established by a minimum of three test points along the test distance.
- For cylinders having length to O.D. ratios larger than 7, the minimum number of test points is four. The test points are nearly equally spaced along the test distance. One calibration hole is located at a test distance equal to L/2.

5.3.2 Pressure/Temperature Limits

This subsection is written in the format and discusses the information presented in SRP 5.3.2 Draft Rev. 2.

The regulations requiring the imposition of pressure-temperature limits on the reactor coolant pressure boundary are the following:

> 10 CFR 50.55a, "Codes and Standards," requires that structures, systems, and components (SSC) be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. In addition, General Design Criterion 1 of Appendix A of 10 CFR Part 50, "Quality Standards and Records," requires that the codes and standards used to assure quality products in keeping with the safety function be identified and evaluated to determine their adequacy.
>
> General Design Criterion 14 of Appendix A of 10 CFR Part 50, "Reactor Coolant Pressure Boundary," requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested in order to have an extremely low probability of abnormal leakage, of rapid failure, and of gross rupture. Likewise, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," requires, in part, that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance and testing, and postulated accident conditions, the boundary

behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized. Further, in order to assess the structural integrity of the reactor vessel, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," requires, in part, an appropriate materials surveillance program for the reactor vessel beltline region.

The special requirements regarding susceptibility to pressurized thermal shock (PTS) for reactor vessel beltline materials for Pressurized Water Reactors (PWRs) are not applicable to the ESBWR.

The acceptability of the ESBWR reactor coolant pressure boundary pressure-temperature limits is demonstrated by meeting the relevant requirements of the following Commission regulations:

A. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records," as it relates to quality standards for design, fabrication, erection and testing;

B. 10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary," as it relates to assuring an extremely low probability of abnormal leakage, rapidly propagating failure and gross rupture of the reactor coolant pressure boundary;

C. 10 CFR Part 50, Appendix A, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," as it relates to assuring that the reactor coolant pressure boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized;

D. 10 CFR Part 50, Appendix A, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," as it relates to the reactor vessel materials surveillance program;

E. 10 CFR Part 50, §50.55a, "Codes and Standards", as it relates to quality standards for design, and determination and monitoring of material fracture toughness;

F. 10 CFR Part 50, §50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," as it relates to compliance with the requirements of 10 CFR 50, Appendices G and H;

G. 10 CFR Part 50, §50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," as it relates to fracture toughness criteria for PWRs relevant to pressurized thermal shock events is not applicable to the ESBWR; and

H. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," as it relates to material testing and fracture toughness.

The specific criteria which meet the relevant requirements are as presented in the following subsections.

5.3.2.1 Limit Curves

The pressure/temperature limit curves in Figures 5.3-1 and 5.3-2 are based on the requirements of 10 CFR 50 Appendix G and Regulatory Guide 1.99.

The vessel flange, RPV head and flange areas, feedwater nozzles, bottom head and the core beltline areas were evaluated, and the operating limit curves are based on the most limiting locations. The pressure/temperature limits are based on flaw sizes specified in Paragraph G-2120 of ASME Section III, Appendix G. The maximum through wall temperature gradient from continuous heating or cooling at 55.6°C (100°F) per hour was considered. The safety factors applied were as specified in ASME Section III, Appendix G.

The RT_{NDT} of the vessel materials are determined in accordance with the ASME Section III, Subsection NB-2320, and the requirements are listed in Table 5.3-1.

Temperature Limits for Boltup

Minimum flange and fastener temperatures of RT_{NDT} plus 33°C (60°F) are required for tensioning at preload condition and during detensioning. AS shown in Table 5.3-1, this is higher than that calculated in accordance with the methods described in ASME Section III, Appendix G.

Temperature Limits for ISI Hydrostatic and Leak Pressure Tests

Pressure versus temperature limits for preservice and inservice tests when the core is not critical are shown in Figure 5.3-1.

Operating Limits During Heatup, Cooldown, and Core Operation

Figure 5.3-2 specifies limits applicable for normal reactor operation, including anticipated operational occurrences.

Reactor Vessel Annealing

In-place annealing of the reactor vessel, because of radiation embrittlement, is not necessary because the vessel is predicted to maintain an equivalent safety margin in accordance with the procedures of 10 CFR 50 Appendix G, Paragraph IVA.

Predicted Shift in RT_{NDT} and Drop in Upper-Shelf Energy

For design purposes, the adjusted reference nil ductility temperature and drop in the USE for the ESBWR vessel is predicted in accordance with the requirements of Regulatory Guide 1.99.

The calculations are based on the limits specified in Table 5.3-1 on copper and nickel in the weld and forging material.

The fluence analysis was performed using the NRC accepted methodology documented in Reference 5.3-1. The estimated peak fluence for the vessel base material (1/4 T) and the weld above the TAF (at the inside of the RPV) are provided in Table 5.3-4.

As required by 10 CFR 50 Appendix H, a surveillance program will be conducted in accordance with the requirements of ASTM E-185. The surveillance program will include samples of base metal, weld metal and HAZ material of the beltline forging. Subsection 5.3.1.6 provides additional detail on the surveillance program.

5.3.2.2 Operating Procedures

A comparison of the pressure versus temperature limit in Subsection 5.3.2.1 with intended normal operation procedures of the most severe service level B transient shows that those limits are not exceeded during any foreseeable upset condition. Reactor operating procedures are established so that actual transients would not be more severe than those for which the vessel design adequacy has been demonstrated. Of the design transients, the service level B condition producing the most adverse temperature and pressure condition anywhere in the vessel head and/or shell areas yields a minimum fluid temperature and a maximum peak gauge pressure. Scram automatically occurs as a result of this event prior to a possible reduction in fluid temperature. Figure 5.3-2 shows the temperature required to maintain the vessel gauge pressure within the calculated margin against nonductile failure.

5.3.3 Reactor Vessel Integrity

In accordance with SRP 5.3.3, Draft Revision 2, the portions of the DCD listed below are all related to the integrity of the reactor vessel. Although most of these areas are developed separately in other DCD subsections, the integrity of the reactor vessel is of such importance that a special summary discussion of all factors relating to the integrity of the reactor vessel is warranted. The information in each area is discussed to ensure that the information is complete, and that no inconsistencies in information or requirements exist that would reduce the certainty of vessel integrity.

1. Design

   The basic design of the reactor vessel concerning compatibility of design with established quality standards for material properties and fabrication methods is described in Subsection 5.3.1, "Reactor Vessel Materials," establishes compatibility with required inspections as described in Subsection Subsection 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing."

2. Materials of Construction

   The materials of construction are each taken into consideration as described in Subsection 5.2.3, "Reactor Coolant Pressure Boundary Materials," and in Subsection 5.3.1, "Reactor Vessel Materials."

3. Fabrication Methods

   The processes used to fabricate the reactor vessel, including forming, welding, cladding, and machining, are described in Subsection 5.3.1.

4. Inspection Requirements

   The inspection test methods and requirements are described in Subsection 5.3.1.

5. Shipment and Installation

   Protective measures taken during shipment of the reactor vessel and its installation at the site verify that the as-built characteristics of the reactor vessel are not degraded by improper handling.

6. Operating Conditions

   All the operating conditions as they relate to the integrity of the reactor vessel are

   described in Subsection 5.3.2, "Pressure-Temperature Limits."

7. Inservice Surveillance

Plans and provisions for inservice surveillance of the reactor vessel are described in Subsections 5.3.1 and 5.2.4.

The basic acceptance criteria for each review area are covered by other subsections, so they are discussed here only in general terms. References are made to the subsections that include detailed criteria. The acceptance criteria in these subsections describe methods that meet the requirements of the following Commission regulations in 10 CFR Part 50:

General Design Criteria 1, 4, 14, 30, 31, and 32 of Appendix A; Appendix B; 10 CFR 50.60 and associated Appendices G, and H; and 10 CFR 50.55a.

The specific criteria which meet the relevant requirements are as presented in the following subsections.

The reactor vessel materials, equipment, and services associated with the reactor vessel and appurtenances conform to the requirements of the subject design documents. Measures to ensure conformance include (1) provisions for source evaluation and selection, (2) objective evidence of quality furnished, (3) inspection at the vendor source and (4) examination of the completed reactor vessels.

GE provides inspection surveillance of the reactor vessel fabricator in-process manufacturing, fabrication, and testing operations in accordance with the GE quality assurance program and approved inspection procedures. The reactor vessel fabricator is responsible for the first level inspection of manufacturing, fabrication, and testing activities, and GE is responsible for the first level of audit and surveillance inspection.

Adequate documentary evidence that the reactor vessel material, manufacture, testing, and inspection conforms to the specified quality assurance requirements contained in the design documents is available at the fabricator's plant site.

Regulatory Guide 1.2, Thermal Shock to Reactor Pressure Vessels, states that potential RPV brittle fracture, which may result from Emergency Core Cooling System operation, need not be reviewed in individual cases if no significant changes in presently approved core and pressure vessel designs are proposed. If the margin of safety against RPV brittle fracture due to emergency cooling system operation is considered unacceptable, an engineering solution, such as annealing, could be applied to assure adequate recovery of the fracture toughness properties of the vessel material. Regulatory Guide 1.2 requires that engineering solutions be outlined and requires demonstration that the design does not preclude use of the solutions.

An investigation of the structural integrity of boiling water RPVs during a design basis accident (DBA) will be conducted. It will be determined, based on methods of fracture mechanics that no failure of the vessel by brittle fracture as a result of DBA occurs.

The investigation shall include:

- a comprehensive thermal analysis considering the effect of blowdown and the Gravity-Driven Cooling System reflooding;
- a stress analysis considering the effects of pressure, temperature, seismic load, jetload, dead weight, and residual stresses;
- the radiation effect on material toughness (RT_{NDT} shift and critical stress intensity); and

- methods for calculating crack tip stress intensity associated with a nonuniform stress field following the design basis accident.

Appendix G of the ASME Code, Section III shall be applied as a mandatory procedure for demonstrating protection against nonductile failure. The criteria of 10 CFR 50 Appendix G are interpreted as establishing the requirements of annealing. Paragraph IVB requires the vessels to be designed for annealing of the beltline only where the existence of an adequate safety margin cannot be demonstrated in accordance with Paragraph IVA of 10CFR50 Appendix G. The ESBWR vessel is predicted to maintain an adequate safety margin throughout the life of the vessel; therefore, design for annealing is not required.

For further discussion of fracture toughness of the RPV, refer to Subsections 5.3.1.5 and 5.3.2.

5.3.3.1 Design Bases

Safety Design Basis

The reactor vessel and appurtenances are required to withstand different combinations of loadings for loading conditions specified in the design document resulting from operation under normal and abnormal conditions.

To minimize the possibility of brittle fracture of the nuclear system process barrier, the following are required:

- impact properties at temperatures related to vessel operation have been specified for materials used in the reactor vessel;
- expected shifts in transition temperature during design life as a result of environmental conditions, such as neutron flux, are considered in the design and operational limitations assure that NDT temperature shifts are accounted for in reactor operation; and
- operational margins to be observed with regard to the transition temperature are specified for each mode of operation.

Power Generation Design Bases

The power generation design bases of the reactor vessel are:

- develop a simplified system that provides all safety-related functions [i.e., that failure to provide a safety function is incredible (probability of failure is less than 1x10^{-6} per year)];
- develop the ESBWR vessel with a design life of 60 years with a total plant availability of 92% or greater; and
- design the reactor vessel and appurtenances which allows for a suitable program of inspection and surveillance.

5.3.3.2 Description

5.3.3.2.1 Summary Description

Reactor Vessel

The reactor vessel (Figure 5.3-3) is a vertical, cylindrical pressure vessel of welded low alloy steel forging sections. The vessel is designed, fabricated, tested, inspected, and stamped in accordance with ASME Code, Section III, Class 1 requirements. Vessel dimensions are provided in Table 5.3-3.

In addition, the design documents impose additional requirements to ensure integrity and safety of the vessel. Design of the RPV and its support system meets Seismic Category I equipment requirements. The materials used in the RPV are listed in Table 5.2-4.

The cylindrical shell and top and bottom heads of the RPV are fabricated of low alloy steel, the interior of which is clad with stainless steel weld overlay, except for the top head and most nozzles. The main steam and bottom head drain nozzles are clad with stainless steel weld overlay. The bottom head is clad with Ni-Cr-Fe alloy.

A variety of welding processes, such as electroslag, submerged arc, manual welding, automated gas tungsten arc welding etc.; are used for cladding depending upon the location and configuration of the item in the vessel. Cladding in the "as-clad" condition may be acceptable for some deposits made with automatic processes such as submerged arc welding, gas tungsten arc welding, and electroslag welding. For other processes, particularly where manual welding is employed, some grinding or machining is required. Workmanship samples are prepared for each welding process in the "as-clad" condition and for typically ground surfaces.

The welding material used for cladding in the shell area is ASME SFA 5.9 or SFA 5.4, type 309L or 309MoL for the first layer, and type 308L or 309L/MoL for subsequent layers. For the bottom head cladding, the welding material is ASME SFA 5.14, type ERNiCr-3.

Quality control methods used during the fabrication and assembly of the reactor vessel and appurtenances assure that design specifications are met.

The vessel head is secured to the reactor vessel by studs and nuts. These nuts are tightened with a stud tensioner. The vessel flanges are sealed with two concentric metal seal rings designed to permit no detectable leakage through the inner or outer seal at any operating condition, including heating to operating pressure and temperature at a maximum rate of 55.6°C (100°F) in any one-hour period. To detect seal failure, a vent tap is located between the two seal rings. A monitor line is attached to the tap to provide an indication of leakage from the inner seal ring seal.

Shroud Support

The shroud support brackets (Figure 5.3-3) are welded to the inside of the vessel and are made of Ni-Cr-Fe conforming to ASME Code Case N-580-1. The shroud support brackets support the weight of the steam separators, chimney, top guide, shroud, core plate and the peripheral fuel bundles. The shroud brackets are classified as core support structures and are designed in accordance with the ASME Section III, Subsection NG.

Protection of Closure Studs

BWRs do not use borated water for reactivity control during normal operation. This topic is therefore not applicable.

5.3.3.2.2 Reactor Vessel Design Data

The reactor vessel design pressure, design temperature and hydrostatic test pressure are presented in Table 5.3-1.

Vessel Support

The vessel support (Figure 5.3-3) is considered a sliding support block type as defined in ASME Code, Section III, Subsection NF-3124. Sliding supports are provided at a number of positions around the periphery of the vessel. One end of each sliding support is fastened to a circumferential RPV flange segment that is forged integral to the vessel shell ring at that RPV elevation. The other end of each sliding block is restrained by sets of steel guide blocks that are attached to the reactor pedestal support brackets. Under this configuration, each sliding support is relatively free to expand in the radial direction but is restrained in the vertical and vessel tangential directions.

The vessel supports are constructed of low alloy or carbon steel. The vessel support is designed to withstand the loading conditions specified in the design documents and meet the stress criteria of ASME Code, Section III, Subsection NF.

Control Rod Drive Housings

The control rod drive (CRD) housings are inserted through the CRD penetrations in the reactor vessel bottom head and are welded to forged stub tubes made of Ni-Cr-Fe ASME Code Case N-580-1 material. Each housing transmits loads through the stub tubes to the bottom head of the reactor. These loads include the weights of a control rod, a control rod drive, a control rod guide tube (CRGT), an orificed fuel support, and the four fuel assemblies that rest on the orificed fuel support. The housings are provided with lateral supports and are fabricated of low carbon austenitic stainless steel and designed in accordance with ASME Section III, Subsection NB for the pressure boundary portion of the housing and in accordance with ASME Section III, Subsection NG for the non pressure boundary portion.

In-Core Neutron Flux Monitor Housings

Each in-core neutron flux monitor housing is inserted though the in-core penetrations in the bottom head and welded to forged Ni-Cr-Fe ASME Code Case N-580-1 stub tubes and provided with lateral supports.

An in-core flux monitor guide tube is welded to the top of each housing and a startup range neutron monitor (SRNM) or a local power range monitor (LPRM) is bolted to the seal/ring flange at the bottom of the housing outside the vessel. The housings are fabricated of low carbon austenitic stainless steel and are designed in accordance with ASME Section III, Subsection NB.

Reactor Vessel Insulation

The reactor pressure vessel (RPV) insulation is reflective metal type, constructed entirely of series 300 stainless steel and designed for a 60-year life. The insulation is made of prefabricated units engineered to fit together and maintain the insulation efficiency during temperature changes. The insulation is designed to remain in place and resist damage during a safe shutdown earthquake. Each unit is designed to permit free drainage of any moisture that may accumulate in the unit and prevent internal pressure buildup due to trapped gases.

The insulation for the RPV is supported from the biological shield wall surrounding the vessel and not from the vessel shell. Insulation for the upper head and flange is supported by a steel frame independent of the vessel and piping. During refueling, the support frame along with the top head insulation is removed. The support frame is designed as a Seismic Category I structure. Insulation access panels and insulation around penetrations are designed in sections with quick release latches, which provide for ease of installation and removal for vessel inservice inspection and maintenance operations. Each insulation unit has lifting fittings attached to facilitate removal. Insulation units attached to the shield wall are not required to be readily removable except around penetrations. The insulation characteristics at operating conditions are as presented in Table 5.3-1.

Reactor Vessel Nozzles

All piping connected to the reactor vessel nozzles has been designed not to exceed the allowable loads on any nozzle. Four drain nozzles are provided in the bottom head. The feedwater inlet nozzles and isolation condenser return nozzles have thermal sleeves. Nozzles connecting to stainless steel piping have safe ends or extensions made of stainless steel. These safe ends or extensions are to be welded to the nozzles after the pressure vessel is heat treated to avoid furnace sensitization of the stainless steel. All nozzles are low alloy steel forgings in accordance with ASME SA-508, Grade 3, Class 1 material; except, the drain nozzles and the water level instrumentation nozzles. The safe end materials used are compatible with the material of the mating pipes. The design of the nozzles is in accordance with ASME Section III, Subsection NB and meet the applicable requirements of the vessel design documents.

Materials and Inspections

The reactor vessel is designed and fabricated in accordance with the applicable ASME Code as defined in Subsection 5.2.1 of this report. Table 5.2-4 defines the materials and specifications. Subsection 5.3.1.6 defines the compliance with reactor vessel material surveillance program requirements.

5.3.3.3 Materials of Construction

All material used in the construction of the RPV conforms to the requirements of ASME Code, Section II materials. In addition, the materials used in the reactor vessel meet the requirements of the design documents to improve the quality of the materials. The vessel heads, shells, flanges, and major nozzles are fabricated from low-alloy steel purchased in accordance with ASME Specifications SA-533 Type B, Class 1 and SA-508, Grade 3, Class 1. Interior surfaces of the vessel are clad with austenitic stainless steel or Ni-Cr-Fe weld overlay. The beltline region is a single forging made of SA-508, Grade 3, Class 1 material. The RPV head fasteners are described in Subsections 5.3.1.7 and 5.3.1.8.

These construction materials were selected because they provide adequate strength, fracture toughness, fabricability, and compatibility with the BWR environment. Their suitability has been demonstrated by long-term successful operating experience in reactor service.

Fabrication Methods

The RPV is a vertical cylindrical pressure vessel of welded construction fabricated in accordance with ASME Code, Section III, Class 1 requirements. All fabrication of the RPV is performed in accordance with GE-approved drawings, fabrication procedures, and test procedures. The shell, RPV head, flanges and major nozzles are fabricated from low-alloy steel forgings or plates. The shell forgings are joined by circumferential welds only. The length of the shell forgings is chosen so as to minimize the number of circumferential welds. Welding performed to join these vessel components is in accordance with procedures qualified to ASME Section III and IX requirements. Weld test samples were required for each procedure for major vessel full penetration welds.

Submerged arc, gas metal arc, gas tungsten arc and manual stick electrode welding processes are employed. Electroslag welding is not used except for cladding. Preheat and interpass temperatures employed for welding of low-alloy steel meet or exceed the requirements of ASME Section III, Appendix D. Post-weld heat treatment of low alloy welds is presented in Table 5.3-1.

Other fabrication processes such as cutting, bending and forming, are performed in accordance with the vessel design documents.

All previous BWR pressure vessels have employed similar fabrication methods. These vessels have operated for many years and their service history is rated excellent.

5.3.3.4 Inspection Requirements

All plates, forgings, and bolting are 100% ultrasonically tested and surface examined by magnetic particle methods or liquid penetrant methods as required by ASME Section III, Subsection NB. Welds on the RPV are examined in accordance with methods prescribed and meet the acceptance requirements specified by ASME Section III, Subsection NB. In addition, the pressure-retaining welds are ultrasonically examined using acceptance standards according to ASME Section XI.

5.3.3.5 Shipment and Installation

The completed reactor vessel is given a thorough cleaning and examination prior to shipment. The vessel is tightly sealed for shipment to prevent entry of dirt or moisture. Preparations for shipment are in accordance with detailed written procedures.

Upon arrival at the reactor site, the reactor vessel is examined for evidence of any contamination as a result of damage to shipping covers. Measures are taken during installation to assure that vessel integrity is maintained; for example, access controls are applied to personnel entering the vessel, weather protection is provided, and periodic cleanings are performed.

5.3.3.6 Operating Conditions

Procedural controls on plant operation are implemented to hold thermal stresses within acceptable ranges and to meet the pressure/temperature limits of Subsection 5.3.2. A limit on rate of change of reactor coolant temperature is imposed per Table 5.3-1, which assures that the vessel closure, closure studs, vessel support, CRD housing, and stub tube stresses and fatigue usage remain within acceptable limits.

These operational limits, when maintained, ensure that the stress limits within the reactor vessel and its components are within the thermal limits to which the vessel was designed for normal operating conditions. To maintain the integrity of the vessel in the event that these operational limits are exceeded, the reactor vessel has been designed to withstand a limited number of transients caused by operator error. Also, for abnormal operating conditions where safety systems or controls provide an automatic temperature and pressure response in the reactor vessel, the reactor vessel integrity is maintained, because the severest anticipated transients have been included in the design conditions. Therefore, it is concluded that the vessel integrity shall be maintained during the most severe postulated transients, because all such transients are evaluated in the design of the reactor vessel.

5.3.3.7 In-service Surveillance

In-service inspection of the RPV shall be in accordance with the requirements of ASME Section XI. The vessel will be examined once prior to startup to satisfy the preoperational requirements of IWB-2000 of ASME Section XI. Subsequent inservice inspection will be scheduled and performed in accordance with the requirements of 10 CFR 50.55a, subparagraph (g) as described in Subsection 5.2.4.

The materials surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and thermal environment. Specimens of actual reactor beltline material will be exposed in the reactor vessel and periodically withdrawn for impact testing. Operating procedures will be modified in accordance with test results to assure adequate brittle fracture control.

Material surveillance programs and inservice inspection programs are in accordance with applicable 10 CFR 50 Appendix H and ASME Code requirements and provide assurance that brittle fracture control and pressure vessel integrity will be maintained throughout the service lifetime of the RPV.

5.3.4 COL Information

Fracture Toughness Data

Fracture toughness data based on the limiting reactor vessel materials will be provided by the COL applicant (Subsection 5.3.1.5). Pressure/temperature limit curves for the RPV will also be provided (Subsection 5.3.2).

Materials and Surveillance Capsule

The following will be identified by the COL holder: (1) specific materials in each surveillance capsule; (2) capsule lead factors; (3) withdrawal schedule for each surveillance capsule; (4) neutron fluence to be received by each capsule at the time of its withdrawal; and, (5) vessel end-of-life peak neutron fluence (Subsection 5.3.1.6.4).

5.3.5 References

5.3-1 GE Nuclear Energy, “GE Methodology to RPV Fast Neutron Flux Evaluations,” Licensing Topical Report NEDC-32983P-A, Class III (Proprietary), August 2000, and NEDO-32983-A, Class I (Non-proprietary), December 2001.

Table 5.3-1

Reactor Vessel Controls

Component	Control(s)
Specified limits for RPV materials used in the core beltline region.	0.05% maximum copper, 0.006% maximum phosphorous, 1.0% maximum nickel (forging) and 0.73% nickel (plate) content in the base materials and a 0.05% maximum copper, 1.0% nickel, 0.008% maximum phosphorous, and 0.05% maximum vanadium content in weld materials.
Studs, nuts, and washers for the main closure flange.	ASME SA-540, Grade B23 or Grade B24 having minimum yield strength level of 893 MPa. The maximum measured ultimate tensile strength of the stud bolting materials shall not exceed 1172 Mpa.
RPV post-weld heat treatment of low-alloy steel welds.	593°C (1100°F) minimum and not exceeding 635°C (1175°F) is applied to all low-alloy steel welds in accordance with ASME Code, Subsection NB-4620.
Toughness of all bolting material exceeding one inch diameter.	Minimum of 61 J (45 ft-lbf) Charpy V energy and 0.64 mm (0.025 in.) lateral expansion at the minimum bolt preload temperature.
Reactor Vessel Design Data	The reactor vessel design pressure is 8.62 MPa gauge (1250 psig) and the design temperature is 302°C (575°F). The preservice hydrostatic test pressure is 10.78 Mpa gauge (1563 psig).
The insulation for the bottom head and lower shell course.	Vertical cylindrical panel approximately 75 to 100 mm (3 to 4 in.) thick. This panel extends vertically up to the vessel support. There is also a horizontal panel between 75 to 100 mm (3 to 4 in.) thick, which connects across the bottom of the vertical insulation panels. This panel is penetrated by the CRD housings, in-core housings, and drain lines. These components are not insulated individually.
Average maximum heat transfer rate of the insulation on the shield wall and around the refueling bellows	736.9 kJ/m^2h of outside insulation surface.

Table 5.3-1

Reactor Vessel Controls

Component	Control(s)
The maximum heat transfer rate for insulation on the top head	682.4 kJ/m^2h
Minimum air temperatures outside the vessel and insulation	38°C (100°F), below and outside bottom head insulation; 38°C (100°F), outside the vessel support; and 57°C (135°F), above the top head.
Average rate of change of reactor coolant temperature during normal heatup and cooldown:	Not to exceed 55.6°C (100°F) during any one-hour period.
Initial RT_{NDT}	-20°C (-4°F) for all materials
Minimum boltup temperature	-20°C (-4°F) + 33°C (60°F) = 13°C (56°F)

Table 5.3-2

Predicted Irradiation Effects on Beltline Materials

Parameter	Value
Adjusted reference temperature at end of life for the weld (1/4 T)	< 0°C (32°F)
Adjusted reference temperature at end of life for the vessel beltline forging (1/4 T)	< 17°C (63°F)
Calculated shift in RT_{NDT} for welds	20°C (36°F)
Calculated shift in RT_{NDT} for beltline forging	37°C (67°F)
Predicted drop in upper shelf energy for welds	9 J (7 ft-lb)
Predicted drop in upper shelf energy for beltline forging	20 J (15 ft-lb)
The end-of-life upper shelf energy	> 68 J (50 ft-lbf)

Table 5.3-3

Reactor Pressure Vessel Dimensions

Dimension	Value
Nominal inner diameter	7.112 m (280 in.)
Nominal wall thickness including clad	182 mm (7.17 in.)
Minimum cladding thickness	3.2 mm (0.125 in.)
Nominal height from the inside of the bottom head (elevation zero) to the inside of the top head	27.56 m (90.4 ft)
Bottom of the active fuel location from elevation zero	4405 mm (14.45 ft)
Top of the active fuel location from elevation zero	7453 mm (24.45 ft)

Table 5.3-4

RPV Fluence Analysis Results

Parameter	Value For 60 Yrs (n/cm^2)
Expected peak neutron fluence at the 1/4 T location	$< 1.37 \times 10^{19}$
Estimated ¼ T fluence for the weld above the TAF	$< 4.14 \times 10^{17}$

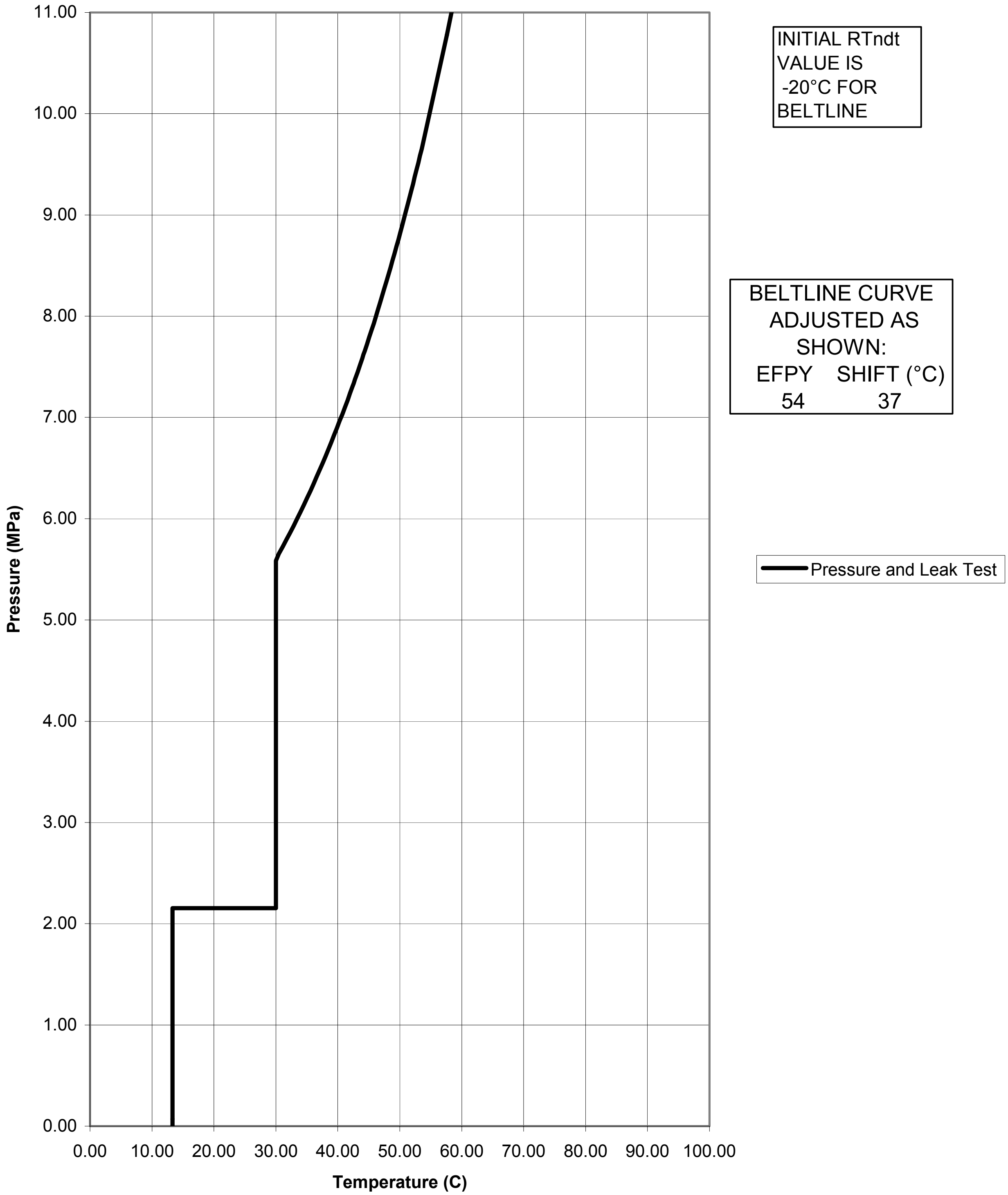


Figure 5.3-1. Minimum Temperatures Required Versus Reactor Pressure for Hydrotest — Core Not Critical

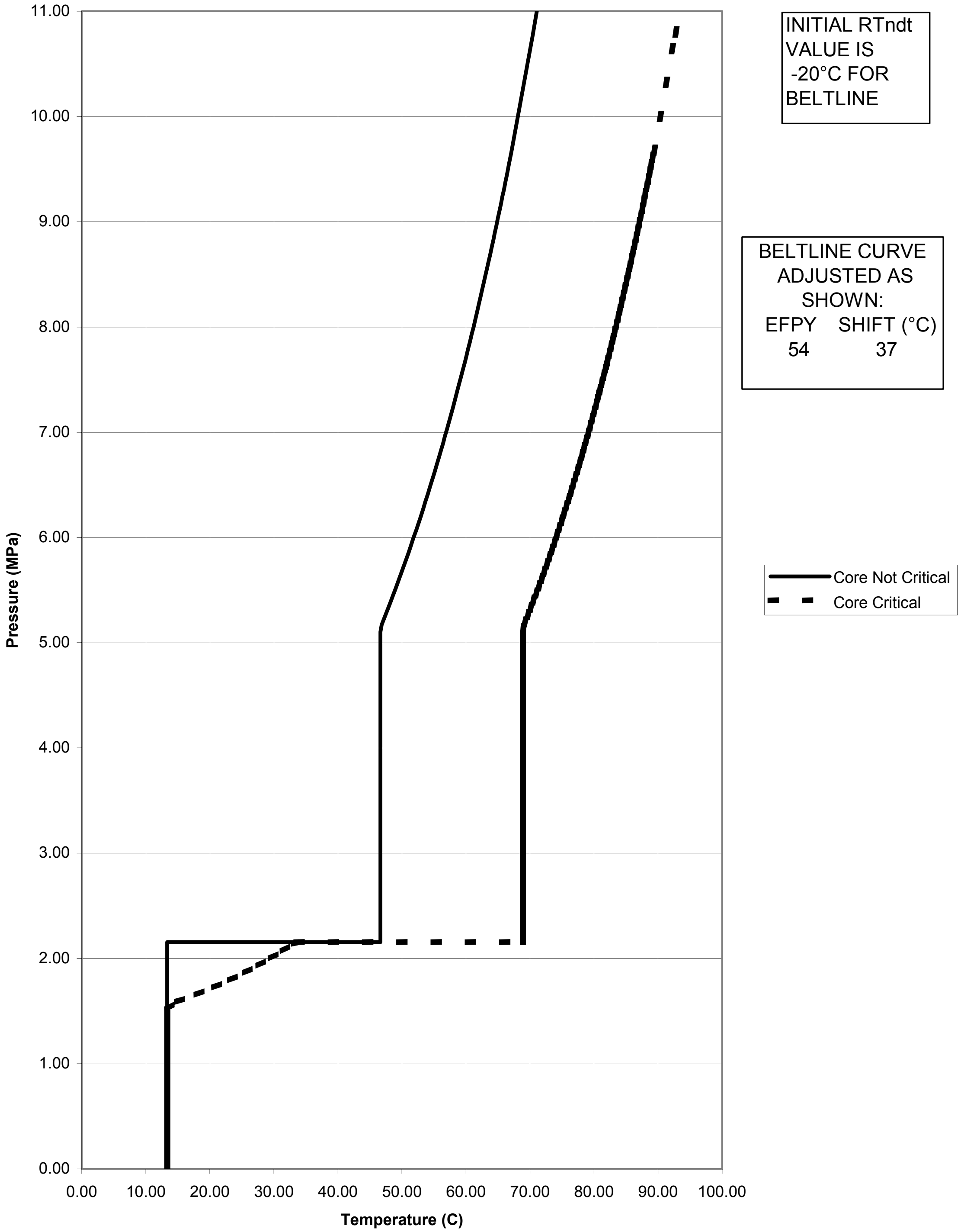


Figure 5.3-2. Minimum Temperatures Required Versus Reactor Pressure for Normal Startup and Shutdown

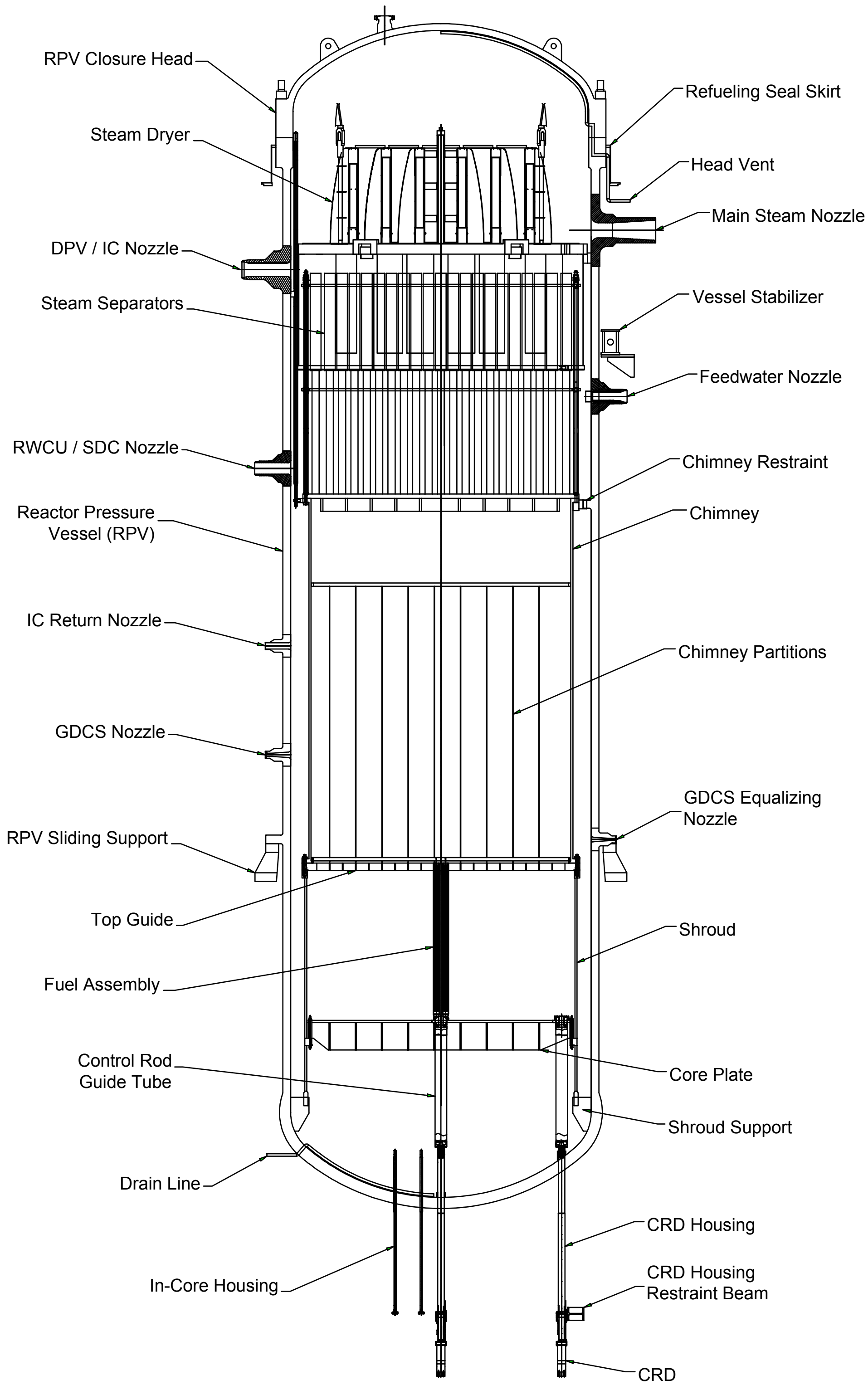


Figure 5.3-3. Reactor Pressure Vessel System Key Features

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 Reactor Recirculation System

The ESBWR relies on buoyancy forces within the reactor vessel to cause recirculation of reactor coolant through the core. There is no forced recirculation system for the ESBWR. The thermal and hydraulic performance of natural recirculation within the reactor core is discussed in Section 4.4.

5.4.1.1 Pump Flywheel Integrity (PWR)

SRP 5.4.1.1, PUMP FLYWHEEL INTEGRITY (PWR), is not applicable to the ESBWR.

5.4.2 Steam Generators (PWR)

5.4.2.1 Steam Generator Materials

SRP 5.4.2.1, STEAM GENERATOR MATERIALS, is not applicable to the ESBWR.

5.4.2.2 Steam Generator Tube Inservice Inspection

SRP 5.4.2.2, STEAM GENERATOR TUBE INSERVICE INSPECTION, is not applicable to the ESBWR.

5.4.3 Reactor Coolant Piping

Because the ESBWR relies on natural circulation within the RPV, no major external reactor coolant piping is connected to the ESBWR pressure vessel.

5.4.4 Main Steamline Flow Restrictors

5.4.4.1 Safety Design Bases

The main steamline flow restrictors are designed to:

- Limit the loss of coolant from the reactor vessel following a steamline rupture outside the containment;
- Withstand the maximum pressure difference expected across the restrictor following complete severance of a main steamline;
- Limit the amount of radiological release outside of the drywell prior to MSIV closure;
- Provide trip signals for MSIV closure.

5.4.4.2 Description

A main steamline flow restrictor (Figure 5.4-1) is provided for each of the four main steamlines with the inside bore of each RPV steam outlet nozzle having the shape of a flow restricting venturi.

The main steamline flow restrictor limits the coolant blowdown rate from the reactor vessel in the event a main steamline break occurs outside the containment (see Table 5.4-1). The flow restrictor is designed and fabricated in accordance with the ASME Code and designed in accordance with the ASME Fluid Meters handbook.

The flow restrictor has no moving parts. Its mechanical structure can withstand the velocities and forces associated with a main steamline break. The flow restrictor design limits flow during a main steamline break to twice the normal full power flow.

The main steamline flow restrictor design substantially limits the steam flow in a severed line, yet it results in negligible increase in steam moisture content during normal operation. The restrictor is also used to measure steam flow to initiate closure of the MSIVs when the steam flow exceeds preselected operational limits. The vessel dome pressure and the venturi throat pressure are used as the high and low pressure sensing locations.

5.4.4.3 Safety Evaluation

In the event that a main steamline breaks outside the containment, the critical flow phenomenon restricts the steam flow rate in the venturi throat (see Table 5.4-1). Prior to isolation valve closure, the total coolant losses from the vessel are not sufficient to produce excessive offsite radiation dose release.

Analysis of the steamline rupture accident (Section 15.4) shows that the core remains covered with water and that the amount of radioactive materials released to the environs through the main steamline break does not exceed the guideline values of published regulations.

The main steamline flow restrictor is Type 308 weld overlay clad. This is similar to the Type 304 cast stainless steel used in previous flow restrictors. It has excellent resistance to erosion/corrosion in a high velocity steam atmosphere. The excellent performance of stainless steel in high velocity steam appears to be due to its resistance to corrosion. A protective surface film forms on the stainless steel, which prevents any surface attack and the steam does not remove this film.

Hardness has no significant effect on erosion/corrosion. For example, hardened carbon steel or alloy steel erodes in applications where soft stainless steel is unaffected.

Surface finish has a minor effect on erosion/corrosion. If very rough surfaces are exposed, the protruding ridges or points erode more rapidly than a smooth surface. Experience shows that a machined or a ground surface is sufficiently smooth and that no detrimental erosion occurs.

5.4.4.4 Inspection and Testing

Because the flow restrictor forms a permanent part of the RPV steam outlet nozzle and has no moving components, no testing program beyond the RPV inservice inspection is planned.

5.4.4.5 Instrumentation Requirements

None

5.4.5 Main Steamline Isolation System

5.4.5.1 Design Bases

Safety Design Bases

The Main Steamline Isolation System shall:

- Isolate the main steamlines within the time established by design basis accident analyses and under the worst-case pressure and flow conditions postulated in the analyses;
- Isolate the main steamlines slowly enough so that simultaneous closure of all steamlines does not induce transients that exceed the nuclear system design limits;
- Isolate each main steamline despite a single failure in either a main steamline isolation valve (MSIV) or in its associated controls;
- Use local stored energy (pneumatic pressure and springs) to close the MSIVs without relying on electrical power as the motive force;
- Isolate the steamlines at full reactor pressure with pneumatic pressure and springs as the motive force;
- Isolate the main steamlines either during or after seismic, hydrodynamic or safety/relief valve blowdown loadings;
- Be testable during normal operating conditions; and
- Isolate the main steamlines for 100 days following a design basis accident.

Power Generation Design Bases

The main steamline isolation system shall:

- Open the main steamlines against a specified maximum differential pressure to permit pressurizing downstream piping during reactor startup;
- Allow steam flow to be achieved without exceeding a design pressure drop; and
- Be designed so an MSIV remains open if one of its two solenoid-operated pilot valves fails.

5.4.5.2 System Description

Summary Description

The main steamline isolation system is a fail-safe system, which isolates the main steamlines during normal, upset, and accident conditions under the full range of reactor pressures and flow conditions. The system consists of eight MSIVs (two per steamline), pneumatic accumulators and connecting piping, and associated controls. The MSIVs are welded in horizontal runs of the steamline, with one valve inside the drywell and the other outside the containment on each line. The system is shown schematically as part of the Nuclear Boiler System in Figure 5.1-2.

Detailed System Description

Figure 5.4-2 shows an MSIV schematically, and the MSIV characteristics are presented in Table 5.4-1. The MSIVs are Y-pattern globe valves installed so that normal steam flow tends to close the valve and increased inlet pressure tends to hold the valve closed. The Y-pattern also streamlines the inlet and outlet valve passages, thereby minimizing the pressure-drop and helping prevent accumulation of debris. The main disk or poppet is attached to the lower end of the stem. The bottom end of the valve stem closes a small pressure-balancing hole in the poppet.

When the hole is open, it acts as a pilot valve to relieve differential pressure forces on the poppet. Valve stem travel is sufficient to produce flow areas past the wide-open poppet greater than the seat port area.

The valve stem penetrates the bonnet flange through replaceable packing. With some packing designs, a leak-off connection may be provided.

The valve is opened by pneumatic pressure, and closed by pneumatic pressure and compressed springs. Helical springs around the spring guide shafts apply force in the direction of valve closure.

Attached to the upper end of the stem is a pneumatic cylinder that opens and closes the valve and a hydraulic dashpot that controls its speed. The MSIV closing speed is adjusted by a valve in the hydraulic return line that bypasses the dashpot piston (see Table 5.4-1). The pneumatic cylinder is supported on the valve bonnet by actuator support and spring guide shafts. The valve control unit is attached to the pneumatic cylinder.

The MSIV valve operator has two closing speeds (see Table 5.4-1). The valve pilot system and accumulator are connected so that when one or both pilots are energized, the accumulator pressurizes the valve operator to open the MSIV. When both pilots are de-energized, the accumulator pressure is switched to pressurize the opposite side of the valve operator and helps the spring close the valve. If one pilot de-energizes (e.g., solenoid failure), the MSIV does not close, and plant operation is not interrupted. A separate solenoid-operated pilot valve with an independent switch is provided for remote-manual testing of the valve.

The MSIVs are designed to a pressure and temperature consistent with the RPV maximum design conditions. Each MSIV is designed to accommodate saturated steam at plant operating conditions. The MSIVs, accumulators, connecting piping up to the check valve from the instrument air or high pressure nitrogen supply, and associated supports are designed to Seismic Category I requirements. The MSIVs form part of the reactor coolant pressure boundary (RCPB) and are therefore Quality Group A and designed and fabricated to ASME Code Section III, Class 1 requirements. The safety-related portions of the pneumatic accumulator and interconnecting piping are Quality Group C and designed to ASME Code Section III, Class 3 requirements.

The MSIVs are designed for a minimum life at the specified operating conditions. In addition to minimum wall thickness required for the design pressure, a corrosion allowance is added for the minimum design life (see Table 5.4-1).

The MSIVs are designed to close under peak accident environmental radiation, pressure, and temperature conditions. In addition, they are designed to remain closed under long-term post-accident environmental conditions (see Table 5.4-1).

The MSIVs located inside containment are normally operated by nitrogen supplied by the High Pressure Nitrogen Supply System (HPNSS) (Subsection 9.3.8) to avoid introducing oxygen into the inerted containment. If the HPNSS is not available, the inboard MSIVs can be operated with compressed air from the Instrument Air System (IAS). The MSIVs located outside the containment are operated by compressed air supplied by the IAS.

An accumulator assists MSIV closure when the make-up pneumatic supply is not available. A check valve in the nitrogen/air supply line prevents loss of accumulator pressure if the supply pressure decreases (e.g., rupture of the line). The accumulator is sized for at least one valve closure.

System Operation

The MSIVs are remote-manually operated from the main control room. Each valve is individually controllable. During normal plant operation, the MSIVs can be tested by cycling them in the slow closing speed (this may require reduction in reactor power to maintain steam flow and pressure within limits). Once initiated, the test sequence is automatic. After normal plant shutdown, the valves can be closed with remote manual switches.

The MSIVs close at the fast speed on various automatic signals indicating abnormal plant conditions, including:

- Reactor low water level;
- Main steamline high flow;
- Low turbine inlet pressure;
- Main steamline tunnel (outside containment) high ambient temperature;
- Low condenser vacuum (unless procedurally bypassed); and
- Turbine building high main steamline ambient temperature.

In the most demanding case (a main steamline rupture downstream of an MSIV), steam flow quickly increases until a venturi flow restrictor installed in each main steamline reactor vessel nozzle prevents further increase. During the major part of valve closure travel the MSIV has little effect on flow reduction because the venturi restrictor chokes the flow. After the valve is nearly closed, steam flow is reduced as a function of the valve area versus travel characteristic.

5.4.5.3 Safety Evaluation

The main steamline isolation system is designed to accomplish the following safety-related functions:

- Limit the loss of reactor coolant in the event of a main steamline break;
- Limit the release of normal reactor coolant radioactivity to the environment in the event of a main steamline break; and
- Help maintain long-term containment leaktightness for accidents in which a significant radioactive release from the reactor core is postulated.

The analysis of a complete, sudden steamline break outside the containment is described in Section 15.4. The analysis shows that the fuel barrier is protected against loss of cooling if the MSIVs close within the longest design closing time plus instrumentation (closing signal) delay (see Table 5.4-1). The calculated radiological effects of the radioactive material assumed to be released with the steam are well within the guideline values.

The analyses of other loss-of-coolant accidents in which large radioactive source terms are postulated are also discussed in Section 15.4. These analyses demonstrate that acceptable off-site dose consequences are maintained when containment leaktightness is maintained, including the specified MSIV leaktightness, and assuming failure of one MSIV to close.

The shortest design closing time of the MSIVs is also shown to be satisfactory (see Table 5.4-1). In the limiting transient [MSIV closure with failure of direct scram (i.e., scram occurs on high neutron flux rather than MSIV position)] reactor vessel design limits are not exceeded (Subsection 5.2.2).

The ability of the MSIV to close in a few seconds after a steamline break, under conditions of high pressure differentials and flow rates and with flow mixtures ranging from mostly steam to mostly water has been demonstrated in a series of dynamic tests. A 500 mm (20-inch) nominal diameter valve was tested in a range of steam-water blowdown conditions simulating postulated accident conditions (Reference 5.4-1).

Two isolation valves provide redundancy so either can perform the isolation function. This also permits leak-testing either valve during shutdown after the other valve is closed. The inside valve, outside valve, and respective control systems are physically separated.

Electrical equipment associated with the isolation valves and operated in an accident environment are limited to the wiring, solenoid valves, junction boxes, electrical connectors, and position switches on the isolation valves.

5.4.5.4 Testing and Inspection Requirements

During fabrication, the following tests (among others) are performed to assure that the MSIVs function as designed:

- Verification of MSIV closing speeds (see Table 5.4-1) - each valve is tested at rated pressure and no flow;
- Seat leakage measurements - both water leakage and air leakage; and
- Hydrostatic testing and nondestructive examinations per ASME Code requirements.

After installation, preoperational testing (described in Section 14.2) assures that the MSIVs will operate as designed, including opening and closing speeds, leaktightness, generation of indication and trip signals, and response to actuation logic signals. The pneumatic operating system is also functionally tested and tested for leaktightness.

The MSIVs are tested for operability during plant operation and planned outages. During outages, the MSIVs are functionally tested, leak-tested, and visually inspected. Leak-testing provisions are further discussed in Subsection 6.2.6. Required periodic tests and inspections of the MSIVs are identified in the plant-specific Technical Specifications.

5.4.5.5 Instrumentation Requirements

MSIV position is indicated in the main control room.

Three limit switches are provided on each MSIV. One limit switch is used by the Reactor Protection System (RPS) to initiate scram on valve closure and by the Isolation Condenser System (ICS) for automatic system initiation on valve closure. The second limit switch is used by the Nuclear Boiler System (NBS) for MSIV position indication. The third limit switch is used as a test switch during MSIV testing and is set such that an MSIV closes far enough to actuate the RPS and ICS limit switches to demonstrate their continued operability. The third switch is also used for MSIV position indication and turbine control system trip. MSIV instrumentation requirements are described further in Section 7.3.

5.4.6 Isolation Condenser System (ICS)

The ESBWR Isolation Condenser System is the most comparable system to the BWR Reactor Core Isolation Cooling (RCIC) System discussed in SRP 5.4.6 Draft Rev. 4, which has been used as a guide for this subsection. The ESBWR is a passive plant relying almost exclusively on natural phenomena to drive plant functions which differs significantly from the SRP BWR RCIC which relies heavily on active systems to accomplish its functions. However, the ESBWR Isolation Condenser System does meet the SRP ACCEPTANCE CRITERIA which are based on meeting the relevant requirements of General Design Criteria 4, 5, 29, 33, 34, 54, and 10 CFR 50.63. The specific criteria met by the ESBWR to meet the requirements of the above GDCs and 10 CFR 50.63 are as follows:

A. General Design Criterion (GDC) 4, as related to dynamic effects associated with flow instabilities and loads (e.g. water hammer).

B. GDC 5 as it relates to safety-related structures, systems and components not being shared among nuclear power units unless it can be demonstrated that sharing does not impair its ability to perform its safety function.

C. GDC 29 as it relates to the system being designed to have an extremely high probability of performing its safety function in the event of Anticipated Operational Occurrences (AOOs).

D. GDC 33 as it relates to the system capability to provide reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary so the fuel design limits are not exceeded.

E. GDC 34 as it relates to the system design being capable of removing fission product decay heat and other residual heat from the reactor core to preclude fuel damage or reactor coolant pressure boundary overpressurization.

F. GDC 54 as it relates to piping systems penetrating primary containment being provided with leak detection and isolation capabilities.

G. 10 CFR 50, §50.63, "Loss of All Alternating Current Power," as related to design provisions to support the plant's ability to withstand and recover from a Station Black-Out (SBO) of a specified duration.

The ESBWR passive decay heat removal systems (Isolation Condensers) are capable of achieving and maintaining safe stable conditions for at least 72 hours without operator action following non-LOCA events. Operator action is credited after 72 hours to refill Isolation Condenser pools or initiate non-safety shutdown cooling.

5.4.6.1 Safety Design Bases

Functions

The ICS automatically limits the reactor pressure and prevents Safety Relief Valve (SRV) operation when the reactor becomes isolated following scram during power operations. The ICS, together with the water stored in the RPV, conserves sufficient reactor coolant volumes to avoid automatic depressurization caused by low reactor water level.

The ICS removes excess sensible and core decay heat from the reactor, in a passive way and with minimal loss of coolant inventory from the reactor, when the normal heat removal system is unavailable, following any of the following events:

- Sudden reactor isolation from power operating conditions;
- Station blackout (i.e., unavailability of all AC power); and
- Anticipated Transient Without Scram (ATWS).
- Loss of Coolant Accident (LOCA)

The ICS is designed as a safety-related system to remove reactor decay heat following reactor shutdown and isolation. It also prevents unnecessary reactor depressurization and operation of other Engineered Safety Features (ESFs) which can also perform this function.

In the event of a Loss of Coolant Accident (LOCA), the Isolation Condenser system (ICS) provides additional liquid inventory upon opening of the condensate return valves to initiate the system. The IC system also provides reactor with initial depressurization of the reactor before ADS in event of loss of feed water, such that the ADS can take place from a lower water level.

General System Requirements

The ICs are sized to remove post-reactor isolation decay heat with three out of four ICs operating and to reduce reactor pressure and temperature to safe shutdown conditions, with occasional venting of radiolytically generated noncondensable gases to the suppression pool (see Table 5.4-1).

The ICS shall be designed and qualified as a safety-related system.

The ICS provides isolation valves for containment isolation (Subsection 6.2.4).

Performance Requirements

The heat removal capacity of the ICS (with three of four IC trains in service) at reactor pressure with saturated steam is presented in Table 5.4-1. The condensate return valve stroke-open time and logic delay time is presented in Table 5.4-1.

5.4.6.2 System Description

5.4.6.2.1 Summary Description

The ICS consists of four totally independent trains, each containing an isolation condenser (IC) that condenses steam on the tube side and transfers heat to the IC/PCC pool, which is vented to the atmosphere as shown on the ICS schematic (Figure 5.1-3).

The IC, connected by piping to the reactor pressure vessel, is placed at an elevation above the source of steam (vessel) and, when the steam is condensed, the condensate is returned to the vessel via a condensate return pipe.

The steam side connection between the vessel and the IC is normally open and the condensate line is normally closed. This allows the IC and drain piping to fill with condensate, which is maintained at a subcooled temperature by the pool water during normal reactor operation.

The IC is started into operation by opening condensate return valves and draining the condensate to the reactor, thus causing steam from the reactor to fill the tubes which transfer heat to the cooler pool water.

5.4.6.2.2 Detailed System Description

The ICS consists of four high-pressure, totally independent trains, each containing a steam IC as shown on the ICS schematic (Figure 5.1-3).

Each IC is made of two identical modules (see Table 5.4-1). The units are located in a large water pool (IC/PCC pool) positioned above, and outside, the ESBWR containment (drywell).

The IC is configured as follows:

- The steam supply line (properly insulated and enclosed in a guard pipe which penetrates the containment roof slab) is vertical and feeds two horizontal headers through four branch pipes. Each pipe is provided with a built-in flow limiter, sized to allow natural circulation operation of the IC at its maximum heat transfer capacity while addressing the concern of IC breaks downstream of the steam supply pipe. Steam is condensed inside vertical tubes and condensate is collected in two lower headers. Two pipes, one from each lower header, take the condensate to the common drain line which vertically penetrates the containment roof slab.
- A vent line is provided for both upper and lower headers to remove the noncondensable gases away from the unit, during IC operation. The vent lines are routed to the containment through a single penetration.
- A purge line is provided to assure that, during normal plant operation (IC system standby conditions), the excess of hydrogen (from the hydrogen water chemistry control additions) or air from the feedwater does not accumulate in the IC steam supply line, thus assuring that the IC tubes are not be blanketed with noncondensables when the system is first started. The purge line penetrates the containment roof slab.
- Isolation containment valves are provided on the steam supply piping and the condensate return piping.
- Located on the condensate return piping just upstream of the reactor entry point is a loop seal and a parallel-connected pair of valves: (1) a condensate return valve (motor-operated, fail as is) and (2) a condensate return bypass valve (nitrogen piston operated, fail open). These two valves are closed during normal station power operations. Because the steam supply line valves are normally open, condensate forms in the IC and develops a level up to the steam distributor, above the upper headers. To start an IC into operation, the motor-operated condensate return valve or condensate return bypass valve is opened, whereupon the standing condensate drains into the reactor and the steam-water interface in the IC tube bundle moves downward below the lower headers to a point in the main condensate return line. The fail-open nitrogen piston-operated condensate return bypass valve opens if the DC power is lost.

- System controls allow the reactor operator to remote-manually open both of the condensate return valves at any time.
- A loop seal at the RPV condensate return nozzle assures that condensate valves do not have superheated water on one side of the disk and subcooled water on the other side during normal plant operation, thus affecting leakage during system standby conditions. Furthermore, the loop seal assures that steam continues to enter the IC preferentially through the steam riser, irrespective of water level inside the reactor, and does not move counter-current back up the condensate return line.

During ICS normal operation, noncondensable gases collected in the IC are vented from the IC top and bottom headers to the suppression pool. Venting is controlled as follows:

- Two normally closed, fail-closed, solenoid-operated lower header vent valves are located in the vent line from the lower headers. They can be actuated both automatically (when RPV pressure is high and either of condensate return valves is open) and manually by the control room operator. Two normally closed, motor-operated lower header bypass vent valves allow the operator to vent noncondensable gases in case of failure of the automatic lower header vent valves.
- The vent line from the upper headers with two normally closed, fail-closed, solenoid-operated upper header vent valves is provided to permit opening of this noncondensable gas flow path by the operator, if necessary.
- All the vent valves are located in vertical pipe run near the top of the containment. The vent piping is sloped to the suppression pool to prevent accumulation of condensate in the piping.

During ICS standby operation, discharge of hydrogen excess or air is accomplished by a purge line that takes a small stream of gas from the top of the isolation condenser and vents it downstream of the RPV on the main steamline upstream of the MSIVs.

Each IC is located in a subcompartment of the Isolation Condenser/Passive Containment Cooling (IC/PCC) pool, and all pool subcompartments communicate at their lower ends to enable full utilization of the collective water inventory, independent of the operational status of any given IC train. A valve is provided at the bottom of each IC/PCC pool subcompartment that can be closed so the subcompartment can be emptied of water to allow IC maintenance.

Pool water can heat up to about 101°C (214°F); steam formed, being nonradioactive and having a slight positive pressure relative to station ambient, vents from the steam space above each IC segment where it is released to the atmosphere through large-diameter discharge vents.

A moisture separator is installed at the entrance to the discharge vent lines to preclude excessive moisture carryover.

IC/PCC pool makeup clean water supply for replenishing level during normal plant operation is provided from the Fuel and Auxiliary Pools Cooling System (FAPCS) (Subsection 9.1.3).

A safety-related independent FAPCS makeup line is provided to convey emergency makeup water into the IC/PCC pool, from piping connections located at grade level in the reactor yard external to the reactor buildings.

Four radiation monitors are provided in the IC/PCC pool steam atmospheric exhaust passages for each IC train. They are shielded from all radiation sources other than the steam flow in the exhaust passages for a specific IC train. The radiation monitors are used to detect IC train leakage outside the containment. Detection of a low-level leak (radiation level above background - logic 2/4) results in alarms to the operator. At high radiation levels (exceeding site boundary limits - logic 2/4), isolation of the leaking isolation condenser occurs automatically by closure of steam supply and condensate return line isolation valves.

Four sets of differential pressure instrumentation are located on the IC steam line and another four sets on the condensate return line inside the drywell. Detection of excessive flow beyond operational flow rates in the steam supply line or in the condensate return line (2/4 signals) results in alarms to the operator, plus automatic isolation of both steam supply and condensate return lines of the affected IC train.

5.4.6.2.3 System Operation

Normal Plant Operation

During normal plant operation, each IC train is in "ready standby," with both steam supply isolation valves and both isolation valves on the condensate return line in a normally open position, condensate level in the IC extending above upper headers, condensate return valve-pair both closed, and with the small vent lines from the IC top and bottom headers to the suppression pool closed. Steam flow is induced from the steam distributor through the purge line by the pressure differential caused by flow in the main steamline.

The valve status, failure mode, actuation mode, pipe size, valve type and line are shown in Table 5.4-2.

Plant Shutdown Operation

During refueling, the IC is isolated from the reactor, with all steam supply and condensate return isolation valves closed. The IC lower and upper header vent valves are also closed.

Isolation Condenser Operation

Any of the following sets of signals generates an actuation signal for ICS to come into operation:

- Two or more MSIV valve positions at ≤ 92% open, with a MSIV valve position on another MSL ≤ 92% open, with Reactor Mode Switch in "run" only (% open values are those used in the safety analyses);
- RPV dome gauge pressure ≥ 7.447 MPa (1080 psig) for 10 seconds;
- Reactor water level below Level 2, with time delay;
- Reactor water level below Level 1.5;
- Loss of power generation busses;
- Operator remote manual initiation;

When one of these ICS initiation signals occurs, condensate return valves open within required stroke time (Table 5.4-1), which starts IC operation. If, during IC operation and after the initial transient, the RPV pressure increases above 7.516 MPa gauge (1090 psig), the bottom vent valves automatically open; and when the RPV pressure decreases below 7.447 MPa gauge (1080 psig) (reset value) and after a time delay to avoid too many cycles, these valves close.

After reactor isolation and automatic IC System operation, the control room operator can control the venting of noncondensable gases from the IC, to enable it to hold reactor pressure below safe shutdown limits.

The ICS is also designed to provide makeup water to the RPV during LOCA event by draining the IC and condensate return line standby inventory into the RPV. The emergency core cooling system (ECCS) and the ICS are designed to flood the core during a LOCA event to provide required core cooling. By providing core cooling following a LOCA, the ECCS and ICS, in conjunction with the containment, limits the release of radioactive materials to the environment following a LOCA.

5.4.6.3 Safety Evaluation

The Isolation Condenser System is used to transfer decay and residual heat from the reactor after it is shutdown and isolated. This function can also be performed by the RWCU/SDC system or other Engineered Safety Features (ESF) of ADS, PCCS, and GDCS which back up the ICS. The Isolation Condenser System is designed and qualified as a safety-related system to comply with 10 CFR 50 Appendix A, Criterion 34 and to avoid unnecessary use of other ESFs for residual heat removal.

The ICS parts (including isolation valves) which are located inside the containment and out to the IC flow restrictors are designed to ASME Code Section III, Class 1, Regulatory Guide 1.26, Quality Group A. The ICS parts which are located outside the containment downstream of the flow restrictor are designed to ASME Code Section III, Class 2, Regulatory Guide 1.26, Quality Group B. The electrical design systems are designed to comply with Class 1E requirements per Regulatory Guide 1.153, and the entire system is designed to Seismic Category I per Regulatory Guide 1.29.

Three out of four ICS trains remove post-reactor isolation decay heat and depressurize the reactor to safe shutdown conditions when the reactor is isolated after operation at 100% power.

As protection from missile, tornado and wind, the ICS parts outside the containment (the Isolation Condenser itself) are located in a subcompartment of the safety-related IC/PCC pool to comply with 10 CFR 50 Appendix A, Criteria 2, 4 and 5.

The IC steam supply pipes include flow restrictors. The IC condensate drain pipes are of limited area so that, in the event of an IC piping or tube rupture in the IC/PCC pool, the resulting flow-induced dynamic loads and pressure buildup in the IC/PCC pool are limited. Penetration sleeves are used at the locations where the IC steam supply and condensate return pipes enter the pool at the containment pressure boundary. These penetration sleeves are designed and constructed in accordance with the requirements specified in Section 3.6. The ICS valve actuators are to be qualified for service inside the drywell for continuous service under normal conditions and to be operable in a DBA environment. Thereafter, the valves are required to remain in their last position.

The ICS steam supply lines, condensate return lines, instrument lines, and vent lines that penetrate containment are provided with isolation valves to satisfy containment isolation requirements as discussed in Subsections 6.2.4.

Compliance of instrumentation and control equipment is addressed in Subsection 7.4.4.

5.4.6.4 Testing and Inspection Requirements

Inspection

During plant outages, routine ISI is required for the isolation condenser, piping containment penetration sleeves, and supports according to ASME Code Section III and Section XI (requirements for design and accessibility of welds).

IC removal for routine inspection is not required.

Ultrasonic inspection is required for IC tubes/headers welds. IC tubes are inspected by the eddy current method.

Testing

Periodic heat removal capability testing of the ICs is required during plant operation. This test is accomplished using data derived from the temperature sensor located downstream of the condensate return isolation valve, together with the differential pressure signal from one of the dPTs, on the condensate return line.

During normal plant operation, a periodic surveillance test of normally-closed condensate return and condensate return bypass valves on condensate line to RPV, being moved into an open condition, are performed.

The test procedure for the condensate return valves starts after the condensate return line isolation valves are closed; this avoids subjecting the IC to unnecessary thermal heatup/cooldown cycles.

Isolation valves on the steam supply line shall remain open to avoid IC depressurization.

The test is performed by the control room operator via remote manual switches that actuate the isolation valves and the condensate return valves; the opening and closure of the valves is verified by their status light.

The procedure is as follows:

- Close steam supply isolation valves.
- Fully open and subsequently close condensate return and then condensate return bypass valve.
- Reopen isolation valves to put the IC in standby condition.

The isolation valves shall be tested periodically, one at a time.

If a system actuation signal occurs during the test, all the valves automatically align to permit the IC to start operation.

Each vent valve is periodically tested.

The valves which are located in series shall be opened one at a time during normal plant operation. A permissive is provided for that (the operator can open one vent valve if the other one in series is closed).

The purge line root valve is periodically tested.

5.4.6.5 Instrumentation Requirements

Control logic for ICS system is addressed in Subsection 7.4.4. The following paragraphs give a brief description of the instrumentation for each of the IC subsystems shown on Figure 5.1-3.

Four radiation sensors are installed in each IC pool exhaust passages to the outside vent lines that vent the air and evaporated coolant (vapor) to the environment. These sensors are part of the Leak Detection and Isolation System described in Subsection 5.2.5.2. On high radiation signal coming from two of the four radiation monitors installed near each IC compartment, all the lines from/to the IC are isolated. This means closure of all steam supply and condensate return isolation valves. The high radiation can be due to a leak from any IC tube and a subsequent release of noble gas to the air above the IC/PCC pool surface.

Four sets of differential pressure instrumentation on each steam supply line and another four sets on each condensate return line are used to detect a possible LOCA.

High dPT signal, coming from two of four dPT sensors on the same line (steam or condensate), closes all isolation valves and therefore renders the IC inoperable.

The operator cannot override either the high radiation signals from the IC atmosphere vents or the high differential pressure IC isolation signals.

A temperature element is provided in each vent line, downstream of the valves, to confirm vent valve function. These temperature elements send a signal to the control room.

A temperature element is provided in the condensate return line, downstream of isolation valve F004 and at the bottom and top of the condensate line at the RPV connection. Each temperature element is connected to the main control room. These temperature measurements provide information on temperature stratification in the piping.

A temperature element is also provided in the upper part of the IC steam supply line in the drywell that can be used to confirm the steam line is near the steam saturation temperature in the RPV and is therefore largely free of noncondensable gases.

A test connection with an end cap is provided at the upstream side of the outer steam supply isolation valve on the steam supply line, to mount a test pressure indicator and perform leak tests on steam supply isolation valves.

A test connection with an end cap is provided at the downstream side of the outer condensate return isolation valve, on the condensate return line to mount a test pressure indicator and perform leak tests on condensate return isolation valves.

A test connection with an end cap is provided upstream of the motor-operated isolation valve to mount a test pressure indicator and perform leak tests on purge line excess flow valve.

5.4.7 Residual Heat Removal System

SRP 5.4.7 Draft Rev. 4 addresses the Residual Heat Removal (RHR) system as used in previous forced circulation BWR product line plants. The ESBWR is a passive plant and does not have the traditional RHR system. For normal shutdown and cooldown, residual and decay heat is removed via the main condenser and the Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) system as discussed in Subsection 5.4.8. The ICS provides cooling of the reactor when the RCPB becomes isolated following a scram during power operations. The ICS

(Subsection 5.4.6) automatically removes residual and decay heat to limit reactor pressure within safety limits when the reactor isolation occurs.

Additional reactor heat removal capability and cooling is provided by Engineered Safety Features (ESFs). The Automatic Depressurization System (ADS) function of the Nuclear Boiler System depressurizes the reactor should the ICS be unable to maintain coolant level (Subsection 6.3.3). Depressurization allows the Gravity-Driven Cooling System (GDCS) to add cool water to the RPV (Subsection 6.3.2). The GDCS is operational at low reactor vessel pressure following pressure reduction by the LOCA or the ADS.

The systems that deal with accomplishing the RHR function meet the requirements of the following regulations as presented in the referenced subsections as follows:

A. General Design Criterion (GDC) 2 with respect to the seismic design of Systems, Structures and Components (SSCs) whose failure could cause an unacceptable reduction in the capability of the residual heat removal function based on meeting position C-2 of Regulatory Guide 1.29 or its equivalent.

B. GDC 4, as related to dynamic effects associated with flow instabilities and loads (e.g., water hammer).

C. GDC 5, which requires that any sharing among nuclear power units of safety-related SSCs does not significantly impair their safety function.

D. GDC 19 with respect to control room requirements for normal operations and shutdown, and;

E. GDC 34, which specifies requirements for systems for residual heat removal (see Subsection 5.4.6).

F. TMI Action Plan item III.D.1.1 of NUREG-0737, equivalent to 10 CFR 50.34(f)(2)(xxvi) with respect to the provisions for a leakage detection and control program to minimize the leakage from those portions of the systems used for RHR that are outside of the containment that contain or may contain radioactive material following an accident.

5.4.8 Reactor Water Cleanup/Shutdown Cooling System

As discussed in SRP 5.4.8 draft Revision 3, the ESBWR meets the relevant requirements of the following regulations:

A. GDC 1 as it relates to the design of the RWCU and components to standards commensurate with the importance of its safety function.

B. GDC 2 as it relates to the RWCU being able to withstand the effects of natural phenomena.

C. GDC 14 as it relates to ensuring the reactor coolant pressure boundary integrity.

D. GDC 60 as it relates to the capability of the RWCU to control the release of radioactive effluents to the environment.

E. GDC 61 as it relates to designing the RWCU with appropriate confinement.

The Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) system performs two basic functions, reactor water cleanup and shutdown cooling, which include the following major activities:

- Purify the reactor coolant during normal operation and shutdown;
- Supplement reactor cooling when the reactor is at high pressure in the hot standby mode;
- Assist in the control of reactor water level during startup, shutdown, and in the hot standby mode;
- Induce reactor coolant flow from the reactor vessel bottom head to reduce thermal stratification during startup;
- Provide shutdown cooling and cooldown to cold shutdown conditions; and
- Provide heated primary coolant for RPV hydrostatic testing and reactor startup.

The RWCU/SDC system is discussed in further detail in Subsections 5.4.8.1 and 5.4.8.2.

5.4.8.1 Reactor Water Cleanup Function

The reactor water cleanup function is performed by the RWCU/SDC system during startup, normal power generation, cooldown and shutdown.

5.4.8.1.1 Design Bases

Safety Design Bases

The RWCU/SDC system does not perform any safety-related functions. Therefore, the RWCU/SDC system has no safety design bases other than for safety-related containment penetrations and isolation valves, as described in Subsection 6.2.4, and provide instrumentation to detect system pipe break outside the containment as described in Subsection 7.4.3

Power Generation Design Bases

The RWCU/SDC system is designed to:

- Remove solid and dissolved impurities from the reactor coolant and measure the reactor water conductivity during all modes of reactor operation. This is done in accordance with Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors";
- Discharge excess reactor water during startup, shutdown, and hot standby conditions and during refueling to the main condenser or to the radwaste system;
- Minimize Reactor Pressure Vessel (RPV) temperature gradients by enhancing circulation through the bottom head region of the RPV and to reduce core thermal stratification at low power;
- Provide heated primary coolant for RPV hydrostatic tests and reactor startup; and
- Have redundant cleanup capacity with respect to major system components.

5.4.8.1.2 System Description

System Description Summary

A main function of the RWCU/SDC system is to purify the reactor water. The RWCU/SDC system consists of two redundant trains, as shown on the RWCU/SDC system schematic (Figure 5.1-4). The major components of each train are two pumps, with an Adjustable Speed

Drive (ASD), one regenerative heat exchanger (RHX), one non-regenerative heat exchanger (NRHX), and a 100% capacity demineralizer. The electrical power supply to the two trains is from separate electrical divisions.

Detailed System Description

The RWCU/SDC system is comprised of two independent pump-and-purification equipment trains. These trains together provide redundant cleanup capacity such that each pump train and demineralizer is designed to achieve and maintain the reactor water quality within design specifications. The system processes the water in the primary system during all modes of operation including startup, normal power generation, cooldown and shutdown operation. The capacity of each train for reactor water cleanup is 1% of the rated feedwater flow rate. The RWCU/SDC system flow rates and other system capabilities are provided in Table 5.4-3.

During normal plant operation, the RWCU/SDC system continuously recirculates water taking suction from the mid-vessel area of the RPV and from the reactor bottom head and returning via the feedwater line to the RPV. The reactor water is cooled by flowing through the tube side of the Regenerative Heat Exchanger (RHX) and the Non-Regenerative Heat Exchanger (NRHX) before entering the RWCU/SDC pump suction. The pump discharges the flow to the demineralizer for the removal of impurities and returns and reheats the reactor water via the shell side of the RHX.

Each train of the RWCU/SDC system performs the two functions of reactor water cleanup and shutdown cooling with a common piping system. The RWCU/SDC system suction line from reactor bottom up to and including the outboard isolation valve, reactor bottom flow sample line up to and including the outboard isolation valve, pumps, demineralizer, pump suction line including suction valves up to and including the demineralizer downstream isolation valve, demineralizer bypass valve and upstream piping are constructed of stainless steel. The remaining system is constructed of carbon steel..

During reactor startup, while maintaining the flow within the cooling capacity of the NRHX, the flow from the demineralizers can be directed to the main condenser hotwell or the liquid radwaste system low conductivity tank for the removal of reactor water that thermally expands during heatup and for removal of inflow from the Control Rod Drive (CRD) system to the RPV.

For RPV hydrotesting and startup, external heating of the reactor water is required if decay heat is not available or the heatup rate from decay heat would be too slow. Feedwater is used to heat the reactor and reactor water.

System Components

The supply side of the RWCU/SDC system is designed for the RCPB design pressure plus 10%. Downstream of the pumps, the pump shutoff head at 5% overspeed is added to the supply side design pressure.

The RWCU/SDC system includes the following major components:

- Demineralizers;
- Pumps and adjustable speed motor drives;
- Non-regenerative heat exchangers;

- Regenerative heat exchangers; and
- Valves and piping;

Demineralizer — The RWCU/SDC system has a mixed bed demineralizer.

A full shutdown flow bypass line with a flow control valve is provided around each demineralizer unit for bypassing these units whenever necessary.

Resin breakthrough to the reactor is prevented by a strainer in the demineralizer outlet line to catch the resin beads. Non-regeneration type resin beads are used, minimizing the potential for damaged beads passing through the strainer to the reactor. The demineralizer is protected from high pressure differential by a bypass valve. The demineralizer is protected from excessive temperature by automatic controls that first open the demineralizer bypass valve and then close the demineralizer inlet valve.

Resin bed performance is monitored as described in Subsection 9.3.2. When it is desired to replace the resin, the resin vessel is isolated from the rest of the system before resin addition.

Pumps — The RWCU/SDC pumps overcome piping and equipment head losses and feedwater line backpressure and return the treated water to the reactor through the feedwater lines.

The continuous minimum flow rate recommended by the vendor is less than the minimum flow through the pumps during any of the operating modes.

The pumps meets the minimum NPSH requirement for all operating modes

Pumps are protected from damage by foreign objects during initial startup by temporary startup suction strainers.

Adjustable Speed Drive (ASD) — The RWCU/SDC pumps are each powered from ASD. The ASDs receive 480V electrical power at constant ac voltage and frequency. The ASDs convert this to a variable frequency and voltage in accordance with a demand signal. The variable frequency and voltage is supplied to vary the speed of the pump motor. The ASD allows effective control of cooldown rate, and reactor temperature after cooldown.

Regenerative Heat Exchanger—Each RHX is used to recover sensible heat in the reactor water and to reduce the recycle heat loss and avoid excessive thermal stresses and thermal cycles of the feedwater piping. Thermal relief valves are provided on both the shell and tube sides of the RHX.

Non-Regenerative Heat Exchanger—Each NRHX cools the reactor water by transferring heat to the RCCWS.

The maximum allowed cooling water outlet temperature from the NRHX is 60 ^{o} C. Thermal relief valves are provided on the tube side of the NRHX. A shell side relief valve is also provided and is sized on the basis of a tube leakage equivalent to 10% of the tube side flow or to relieve shell side pressure in the event the shell side (cooling water) valves are closed and the tube side flow continues.

Isolation Valves — Only the containment isolation valves and piping perform a safety-related function. Refer to Subsection 6.2.4 for isolation valve descriptions.

Piping — Piping from the RPV to the outboard containment isolation valve forms part of the RCPB and is Quality Group A, ASME Section III, Class 1 and Seismic Category I. Downstream, of the outboard containment isolation valve the piping is Quality Group C, ASME Section III, Class 3, and Seismic Category 1. At the point of introduction of the RWCU/SDC piping to the feedwater lines, the return line of the RWCU/SDC has a thermal sleeve to accommodate (without excessive thermal stresses) the maximum temperature difference that can occur between the two fluid streams under any mode of plant operation. The RWCU/SDC return line from the isolation valve, up to and including the connection to the feedwater line, is Quality Group B, ASME Section III, Class 2, and Seismic Category I.

System Operation

The modes of operation for the cleanup function are described below.

Power Operation — During normal power operation, reactor water flows from the reactor vessel and is cooled while passing through the tube side of the RHXs tube side of the NRHXs. The RWCU/SDC pumps then pump the reactor water through the demineralizers, and back through the RHX shell side where the reactor water is reheated and is returned to the reactor vessel via the feedwater lines.

Startup — During drain and fill operations, the RWCU/SDC system is isolated and depressurized. During draining, the high point vents and low point drains are manually opened. During filling, the low point drains are manually closed and the system is filled with water. Individual high point vents are manually opened to remove any entrapped air.

During heatup, feedwater is introduced in the reactor to raise its temperature, while cold water is overboarded to the main condenser by the RWCU/SDC system. The system is designed to provide sufficient flow through the bottom head connections during heatup, cooldown, and startup operations to prevent thermal stratification and to prevent crud accumulation.

During reactor startup, it is necessary to remove the CRD purge water injected into the RPV and also the excess reactor water volume arising from thermal expansion. The RWCU/SDC system accomplishes these volume removals and thereby maintains proper reactor level until steam can be sent to the main turbine condenser.

After warmup the RPV pressure is brought to saturation by opening the vessel to the main condenser through the main steam and turbine bypass lines to promote deaeration of the reactor water. The RWCU/SDC system normally removes excess water by dumping, or overboarding, to the condenser hotwell. If the demineralizer is bypassed, the radwaste system is used as an alternative flow path to avoid radioactive coolant from entering the condensate system. Overboarding is described in more detail below.

Overboarding — During hot standby and startup, water entering the reactor vessel from the CRD System or water level increase due to thermal expansion during plant heatup, may be dumped, or overboarded, to the main condenser to maintain reactor water level.

Overboarding of reactor water is accomplished by using one of the two system trains for overboarding and the other train for the reactor water cleanup function.

The train in the overboarding mode uses a combination of RWCU/SDC pump flow and pressure control to maintain the reactor water level. A pressure control station is located downstream of the demineralizer. The pressure control station consists of a pressure control valve, a high pressure restriction orifice, an orifice bypass valve, and a main condenser isolation valve.

Reactor water level is automatically controlled by controlling the pump speed and the pressure control valve position through a combination of flow, level, and pressure control signals.

During the early phases of startup, when the reactor pressure is low, the restriction orifice is bypassed. The restriction orifice bypass valve automatically closes when the pressure upstream reaches a predetermined set point to ensure the pressure drop across the pressure control valve and the orifice bypass valve are maintained within their design limits.

During overboarding, the RHX is bypassed and the NRHX is in service to cool the reactor water to minimize two-phase flow in the pressure reducing components and downstream piping. The demineralizer is also in service to ensure the water overboarded to the condenser meets water quality specification requirements. In the event high radiation is detected downstream of the demineralizer, the overboarding flow is manually shifted to the Liquid Waste Management System (LWMS) by first opening the remote manual isolation valve to the radwaste system and then closing the remote manual system isolation valve to the main condenser.

The system piping routed to the main condenser and LWMS is designed with sufficient wall thickness to ensure the stresses are within the stress limits even if subjected to full reactor pressure. Further, the low-pressure portion of the system is protected by the automatic closure of the pressure control valve upon detection of high pressure downstream of the pressure control valve. The system piping routed to the LWMS system is also protected from overpressurization by a pressure relief valve that relieves to the piping routed to the main condenser.

Refueling—During refueling, when the reactor well water may have a stratified layer of hot water on the surface, the RWCU/SDC system can be used to supplement the FAPCS to cool the reactor well water.

5.4.8.1.3 Safety Evaluation

The RWCU/SDC system is classified as a nonsafety-related system except for its RCPB function, containment isolation functions, and providing instrumentation for detection of break in the system outside the containment. Refer to Subsection 6.2.4 for containment isolation valves and to Subsection 7.4.3 for containment isolation and pipe break detection instrumentation.

5.4.8.1.4 Testing and Inspection Requirements

During preoperational testing, system component operability, flow rates, heat removal capacities and controls and interlocks are tested to demonstrate that the RWCU/SDC system meets design requirements.

The functional capabilities of the containment isolation valves are testable in-place in accordance with the inservice inspection requirements. All such leak test connections are isolable by two valves in series. Periodic leak testing of the containment isolation valves is prescribed in the Technical Specifications and described in Subsection 6.2.6.

5.4.8.1.5 Instrumentation

RWCU/SDC system instrumentation is described in Subsection 7.4.3. Measurements for flow rate, pressure, temperature, and conductivity are recorded or indicated in the main control room where suitable alarms are provided. Valves behind shielding are furnished with on-off air operators, individually controlled from a local panel or with extension stems that penetrate the shielding. Section 7.4 provides the RWCU/SDC system interlock schematic which shows control logic for the pumps, isolation valves, and system interlocks.

Flow Measurement

High RWCU/SDC system differential mass flow is detected by density compensated flow signals from the safety-related temperature and flow transmitter that measures the system mass flow from the reactor bottom and the mid-vessel nozzles inside the containment and the safety-related temperature and flow transmitters that measure the mass flow outside the containment in the RWCU/SDC return and overboarding lines. The flow and temperature transmitter signals are sent to the LD&IS in four independent safety-related divisions.

Pump Controls

Each pump is manually operated from the control room by a switch with status indicator.

Each pump is protected from potential cavitation during the shutdown cooling mode by a speed runback set to actuate if the RPV water level falls to Level 3.

The following signals will trip the pump:

- Low Pump Suction Flow
- Low Pump Suction Pressure
- Overspeed

Isolation Valves

Containment isolation valves are both automatically and manually actuated with automatic closure overriding manual opening signals.

The following signals prevent the containment isolation valves from opening, and close them if they are open:

- Leak Detection and Isolation System (LD&IS) signals (see Subsection 7.3.3 for isolation by):
  - Initiation of the Standby Liquid Control (SLC) System.
  - High temperature in main steamline tunnel;
  - Low reactor water level (Level 2); and
  - High RWCU/SDC flow.

SLC System actuation (boron injection) prevents the inboard and outboard isolation valves from opening, or closes them if they are open. This isolation prevents the boron from being removed from the reactor water by the RWCU/SDC system demineralizers.

NRHX High Temperature

Reactor water temperature at the NRHX tube-side outlet is indicated and high-high temperature annunciated in the main control room. This signal will bypass the demineralizer.

System Flow Valves

Each system flow control valve is manually operable from the main control room with indication of position status.

Overboard Flow Control Valves

The valve position of the overboard flow control valve, is controlled from the main control room with a remote manual controller. The control circuits are designed to cause the valve to fail closed and actuate an annunciator.

High pressure downstream and low pressure upstream of the valve automatically close the overboard flow control valve and actuate an annunciator.

Temperature Monitoring

Temperature element is provided on the return lines to feedwater to indicate the return temperature.

To protect the demineralizer resins from high temperature, demineralizer inlet temperature indication and alarms are provided in the main control room.

High temperature activates the alarm to alert the plant operator and automatically isolates the demineralizer and opens the demineralizer bypass.

Conductivity Instrumentation and Sampling Points

The conductivity of the demineralizer influent and effluent process streams is continuously measured and transmitted to recorders in the main control room. Conductivity in excess of water quality requirements is alarmed in the main control room.

A Sampling probe is located in the suction line from reactor bottom in both trains inside the containment. This provides sampling of reactor water during plant operation and post accident sampling of reactor water as well.

Sampling probes are located in the inlet header and in each effluent line of the two demineralizer units. Sample lines from each probe are routed to the sample station.

5.4.8.2 Shutdown Cooling Function

The normal shutdown cooling function is performed by the RWCU/SDC system.

5.4.8.2.1 Design Bases

Safety Design Bases

Refer to Subsection 5.4.8.1.1 for the safety design bases.

Power Generation Bases

The shutdown cooling mode of the RWCU/SDC system shall:

- Remove decay heat during normal plant shutdowns;
- Remove the core decay heat, plus overboard the control rod drive cooling flow after approximately one-half hour following control rod insertion and assuming either the main condenser or ICS is available for initial cooldown; and

- With loss of preferred off-site AC power, bring the plant to cold shutdown in 36 hours in conjunction with the ICS, assuming the most restrictive single active failure.

5.4.8.2.2 System Description

In conjunction with the heat removal capacity of either the main condenser and/or the isolation condensers, the RWCU/SDC system can reduce the RPV pressure and temperature during cooldown operation from the rated design pressure and temperature to below boiling at atmospheric pressure in less than one day (see Table 5.4-3). The system is also designed to control the reactor temperature reduction rate.

The system can be connected to nonsafety-related standby AC power (diesel-generators), allowing it to fulfill its reactor cooling functions during conditions when the preferred power is not available.

The shutdown cooling function of the RWCU/SDC system provides decay heat removal capability at normal reactor operating pressure as well as at lower reactor pressures.

The redundant trains of RWCU/SDC permit shutdown cooling even if one train is out of service; however, cooldown time is extended when using only one train.

In the event of loss of preferred power, the RWCU/SDC system, in conjunction with the isolation condensers, is capable of bringing the RPV to the cold shutdown condition in a day and a half, assuming the most limiting single active failure, and with the isolation condensers remove the initial heat load. Refer to Subsection 5.4.8.1.2 for a description of the RWCU/SDC pump motor adjustable speed drive and its operation for shutdown cooling.

System Operation

The modes of operation of the shutdown cooling function are described below:

Normal Plant Shutdown — The operation of the RWCU/SDC system at high reactor pressure reduces the plant reliance on the main condenser or ICS. The entire cooldown is controlled automatically. As cooldown proceeds and reactor temperatures are reduced, pump speeds are increased and various bypass valves are opened, as described below. During the early phase of shutdown, the RWCU/SDC pumps operate at reduced speed to control the cooldown rate to less than the maximum allowed RPV cooling rate.

In order to maintain less than the maximum allowed RPV cooling rate, both RWCU/SDC trains are placed into operation early during the cooldown, but with the pumps and system configuration aligned to provide a moderate system flow rate. The flow rate for each train is gradually increased as RPV temperature drops. To accomplish this, in each train, the bypass line around the RHX, and the bypass line around the demineralizer are opened to obtain the quantity of system flow required for the ending condition of the shutdown cooling mode. In addition to the RCCWS inlet valve to the NRHX being open, at an appropriate point the motor-operated RCCWS inlet valve opens to increase the cooling water flow to each NRHX.

The automatic reactor temperature control function controls the ASD, controlling the cooldown by gradually increasing the speed of the system pumps up to the maximum pump flow. Water purification operation is continued without interruption.

Over the final part of the cooldown, maximum flow is developed through the RWCU/SDC pumps. After about two weeks, flow rate reduction becomes possible while maintaining reactor coolant temperatures within target temperature ranges.

CRD System flow is maintained to provide makeup water for the reactor coolant volume contraction that occurs as the reactor is cooled down.

The RWCU/SDC system overboarding line is used for fine level control of the RPV water level as needed.

Hot Standby — During hot standby the RWCU/SDC system may be used as required in conjunction with the main or isolation condenser to maintain a nearly constant reactor temperature by processing reactor coolant from the reactor bottom head and the mid-vessel region of the reactor vessel and transferring the decay heat to the RCCWS by operating both RWCU/SDC trains and returning the purified water to the reactor via the feedwater lines.

The pumps and the instrumentation necessary to maintain hot standby conditions are connectable to the Standby AC Power supply during any loss of preferred power.

Refueling — The RWCU/SDC system can be used to supplement the FAPCS spent fuel heat removal capacity during refueling (or other times). It also can provide additional cooling of the reactor well water when the RPV head is off in preparation for removing spent fuel from the core.

Operation Following Transients— In conjunction with the isolation condensers, the system has the capability of removing the core decay heat, plus drain excess makeup due to the CRD purge flow, after one-half hour following control rod insertion.

If the shutdown was caused by an isolation event (which brings the Isolation Condenser System (ICS) immediately into operation when the reactor is in the "run" mode of operation), and assuming the most restrictive single active failure, one or all isolation condensers can be valved-out by the operator in order to provide easier pressure and water level regulation with the RWCU/SDC system.

5.4.8.2.3 Safety Evaluation

The RWCU/SDC system does not perform or ensure any system level safety-related function, and thus, is classified as nonsafety-related.

Refer to Subsection 5.4.8.1.3 for an evaluation of the safety-related containment isolation, and instrumentation for pipe break detection outside the containment functions of the RWCU/SDC system.

5.4.8.2.4 Testing and Inspection Requirements

Refer to Subsection 5.4.8.1.4 for the testing and inspection requirements for the RWCU/SDC system.

5.4.8.2.5 Instrumentation

RWCU/SDC system instrumentation is described in Subsection 7.4.3. The shutdown cooling mode of the RWCU/SDC has an automatic temperature control function that controls the speed of the ASDs to control the coolant temperature as measured by the core inlet thermocouples during the shutdown operation.

Instruments monitoring the temperature of the RCCWS water leaving the NRHX also automatically control the RWCU/SDC system flow by adjusting the pump speed in the event the RCCWS outlet temperature from the NRHX rises above limit.

5.4.9 Main Steamlines and Feedwater Piping

5.4.9.1 Design Bases

Safety Design Bases

The main steam and feedwater lines shall:

- Withstand the stresses from internal pressures, safe shutdown earthquake (SSE) loads, design basis accident loads, hydrodynamic loadings, reactions from discharging safety/relief valves (SRVs) and depressurization valves (DPVs) (for the main steamlines), loads from fast closure of the turbine stop and/or control valves (for the main steamlines), and waterhammer loads (for the feedwater lines);
- Provide for long-term leak-tight isolation of the reactor pressure vessel and the containment.

Power Generation Design Bases

- The main steamlines shall transport steam from the reactor vessel over the full range of reactor power operation and, in conjunction with the main steamline isolation valves (MSIVs), limit the pressure drop from the reactor to the turbine to less than the design value.
- The feedwater lines are designed to supply water to the reactor vessel over the full range of reactor power operation.
- The main steamline piping supports shall permit flooding of the steamlines up to the main turbine stop valves during refueling and other shutdowns without the need for adding temporary supports.

5.4.9.2 Description

The main steamlines consist of carbon steel piping originating at reactor vessel nozzles and running to the main steamline header in the turbine building. From the main steamline header, there are four lines that run to and terminate at the turbine stop valves. The feedwater lines consist of carbon steel piping from the condensate and feedwater system to just inside the steam tunnel and low alloy steel piping from just inside the steam tunnel through containment and then branching to lines connecting to reactor vessel nozzles. The main steam and feedwater piping from the reactor through the isolation valves in the reactor building is shown schematically in Figure 5.4-3. Further descriptions of the main steamlines downstream of the outboard MSIVs and the feedwater lines upstream of the outboard containment isolation valves are contained in Sections 10.3 and 10.4, respectively.

The main steamlines are Quality Group A and ASME Section III, Class 1 from the Reactor Pressure Vessel (RPV) through the outboard MSIVs. They are Seismic Category I from the RPV to the seismic interface restraint downstream of the outboard MSIV. The main steamlines from the outboard MSIV to the turbine stop valves are described in Section 10.3 and Table 3.2-1.

The feedwater lines are Quality Group A and ASME Section III, Class 1 from the RPV through the outboard isolation check valves; Quality Group B and ASME Section III, Class 2 through the motor-operated isolation valves; and Quality Group D, ANSI B31.1 thereafter. They are Seismic Category I from the RPV to the seismic interface restraint upstream of the motor-operated isolation valve and Seismic Category NS thereafter.

Further details on design codes and classifications are provided in Section 3.2 and Table 3.2-1. The design temperature and pressure of the Class 1 portions of the main steam and feedwater lines are the same as that of the RPV (see Table 5.4-1).

Piping and pipe support stress analyses, including assumed load combinations, are discussed in Section 3.9.

The four main steamlines are routed from the reactor vessel nozzles in the upper drywell, through containment penetrations, and through the main steam and feedwater pipe tunnel into the turbine building. Connections from the main steamlines to the safety/relief valves and depressurization valves are located in the upper drywell area. The reactor vessel head vent line is connected to main steamline “A” in the upper drywell. Horizontal process lines are sloped downward in the direction of flow to promote proper drainage.

The two feedwater lines are routed from the turbine building to the main steam and feedwater pipe tunnel, through containment penetrations, and branch to six lines which connect to the RPV in the upper drywell. The use of two lines minimizes the number of containment penetrations while providing two separate flow paths. The six branch lines inside containment provide proper feedwater flow distribution to the RPV. The Control Rod Drive System injection line connects to the Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) System loop “A” return line, which is connected to a thermal sleeve in the “B” feedwater line in the tunnel. The Fuel and Auxiliary Pool Cooling System low pressure coolant injection line connects to the RWCU/SDC system loop “B” return line, which connects to the “A” feedwater line in the tunnel.

A main steamline drain subsystem is provided to drain flooded main steamlines after maintenance, to remove steam condensed during heatup and low power operations, and to provide pressure equalization around the outboard MSIVs during startup. The drain lines are routed to orificed headers, which are connected to the condenser hotwell. The main steamline drain subsystem isolation valves are remote-manually operated from the main control room and are closed when reactor power exceeds a specified power level.

5.4.9.3 Safety Evaluation

Main steam and feedwater line integrity is accomplished by considering all the potential loads in the design, fabrication, installation, testing, and periodic inspection in accordance with the codes and criteria cited in Subsection 5.4.9.2.

The main steamlines are designed to withstand the dynamic loads associated with the various design basis accidents, including a main steamline break outside containment, and external hazard events.

The feedwater lines are designed to withstand the dynamic loads associated with various design basis accidents, including a feedwater line break outside containment, and external hazard events. For the feedwater line break outside containment, the lines are designed to survive the high impact forces that can be generated by rapid closure of the check valves in the line.

5.4.9.4 Testing and Inspection Requirements

Preoperational testing is accomplished as described in Chapter 14. Such testing includes hydrostatic testing for pressure integrity, vibration testing under operating conditions, and flow rate testing.

After commercial operation, inservice inspection is conducted periodically in accordance with the applicable codes to assure continued pressure integrity.

5.4.9.5 Instrumentation Requirements

There is no instrumentation associated with the RCPB portions of the main steamlines or feedwater lines. However, the main steamline excess flow restrictor instrumentation is described in Subsection 5.4.4 and the main steamline isolation system is described in Subsection 5.4.5.

5.4.10 Pressurizer

Not Applicable to the ESBWR.

5.4.11 Pressurizer Relief Discharge System

As stated in SRP 5.4.11, this is a PWR topic. Not Applicable to the ESBWR.

5.4.12 Reactor Coolant System High Point Vents

SRP 5.4.12 addresses ‘Reactor coolant system high point vents provided to exhaust noncondensable gasses from the primary system that could inhibit natural circulation core cooling’. The ESBWR has an RPV head vent system that handles any noncondensable gas buildup at the high point inside the RPV head by sweeping the gasses through a main steamline and then ultimately to the condenser. Additionally, systems that are connected to the RPV and are stagnant during normal plant operation have lines that are sloped to prevent any buildup of noncondensable gasses. The ESBWR features that deal with noncondensable gasses meet the relevant requirements of the following regulations:

A. 10 CFR Part 50.55a and General Design Criteria 1 and 30 as they relate to the vent system components which are part of the reactor coolant pressure boundary being designed, fabricated, erected, and tested and maintained to high quality standards.

B. GDC 14, as it relates to the reactor coolant pressure boundary being designed, fabricated, erected and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

C. 10 CFR 50.46(b) as it relates to the long-term cooling of the core following any calculated successful initial operation of the ECCS to remove decay heat for an extended period of time.

During reactor operation, the noncondensable gases that may collect in the reactor head and the IC steam lines are drawn to the steamline through a vent line from the RPV head and a purge line from each of the ICs. Differential pressure between the reactor head and the downstream steamline location extracts the noncondensables. The noncondensables are swept from these lines to the condenser, where they are extracted. These vents and purge lines are not required to assure natural circulation core cooling.

The vent line used to vent the reactor head noncondensables following a refueling operation is isolated with two normally closed valves during reactor power operation.

The isolation condensers also vent noncondensables to the suppression pool to maintain their performance; however, the ICs are isolable and not part of the primary system. The isolation condenser vents are discussed in Subsection 5.4.6.

5.4.13 Safety/Relief Valves

The ESBWR reactor coolant system has no connected systems that require separate safety-related SRVs for overpressure protection. The safety-related pressure boundaries of systems connected to the RCPB are either protected by the RCPB SRVs, designed to higher pressure than the RCPB, or are open to containment atmosphere so they cannot be overpressurized. See Subsection 5.2.2 for a description of RCPB overpressure protection and SRVs.

5.4.14 Component Supports

Support elements are provided for those components included in the RCPB and the connected systems.

5.4.14.1 Safety Design Bases

Design loading combinations, design procedures, and acceptability criteria are as described in Subsection 3.9.3. Flexibility calculations and seismic analysis for Class 1, 2, and 3 components are confirmed to the appropriate requirements of ASME Code Section III.

Support types and materials used for fabricated support elements conform to Sections NF-2000 and NF-3000 of ASME Code Section III. Pipe support spacing guidelines of Table NF-3611-1 in ASME Code Section III are followed.

5.4.14.2 Description

The use and location of rigid-type supports, variable or constant spring-type supports, snubbers, and anchors or guides are determined by flexibility and seismic/dynamic stress analyses. Direct weldment to thin wall pipe is avoided where possible.

5.4.14.3 Safety Evaluation

The flexibility and seismic/dynamic analyses are performed for the design of adequate component support systems under all loading conditions, including temporary and transient conditions, expected by each component. Provisions are made to provide spring-type supports for the initial dead weight loading due to flooding of steam systems' piping to prevent damage to this type support.

5.4.14.4 Testing and Inspection Requirements

After completion of the installation of a support system, all hangers and snubbers are visually examined to assure that they are in correct adjustment to their cold setting position. Upon hot startup operations, thermal growth is observed to confirm that spring-type hangers and snubbers can function properly between their hot and cold setting positions. Final adjustment capability is provided on all hanger and snubber types.

Weld inspections and standards are in accordance with ASME Code Section III. Welder qualifications and welding procedures are in accordance with ASME Code Section IX and Subsection NF-4300 of ASME Code Section III.

5.4.14.5 Instrumentation Requirements

None.

5.4.15 COL Information

None.

5.4.16 References

5.4-1 General Electric Company, "Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves," APED-5750, March 1969.

Table 5.4-1

Component and Subsystem Design Controls

Component/Subsystem	Control(s)
The main steamline flow restrictor	Limits the coolant blowdown rate from the reactor vessel in the event a main steamline break occurs outside the containment to a (choke) flow rate equal to or less than 200% of rated steam flow at 7.07 MPa gauge (1025 psig) upstream pressure. The throat diameter is ≤ 355 mm.
The ratio of the main steamline flow restrictor venturi throat diameter to steamline inside diameter:	Approximately 0.5, which results in a maximum pressure differential (unrecovered pressure) of about 0.10 MPa (15 psi) at 100% of rated flow. This design limits the steam flow in a severed line to less than 200% rated flow, yet it results in negligible increase in steam moisture content during normal operation.
The main steamline flow restrictor duty:	Exposed to steam of about 0.10% moisture flowing at velocities of 53 m/sec (steam piping ID) to 212 m/sec (steam restrictor throat).
MSIV characteristics:	Nominally 700 mm (28 in.) diameter spring-loaded, pneumatic, piston-operated valves that fail closed on loss of pneumatic pressure to the valve actuator
MSIV rated steam flow:	2.19 x 10^6 kg/hr (4.82 x 10^6 lbm/hr).
MSIV pneumatic cylinder actuator can open the MSIV poppet with a maximum differential pressure of:	1.38 MPa (200 psi) across the isolation valve in a direction that tends to hold the MSIV closed.
The MSIV poppet travels approximately 90% of the valve stem travel to close the main steam port area;	Approximately the last 10% of the valve stem travel closes the pilot valve.
MSIV fast closing speed:	3.0 – 5.0 seconds when operating gas is admitted to the upper piston compartment and the lower piston compartment is vented to the atmosphere.
MSIV slow closing speed:	45 – 60 seconds, by admitting operating gas to both the upper and lower piston compartments.
MSIV steam design envelope:	Designed to accommodate saturated steam at plant operating conditions with moisture content of approximately 0.5%.

Table 5.4-1

Component and Subsystem Design Controls

Component/Subsystem	Control(s)
MSIV design life	60 years service at operating conditions.
MSIV corrosion allowance:	60 years service.
MSIVs are designed to remain closed under long-term post-accident environmental conditions:	≥ 100 days.
MSIV flow established by choked flow at the venturi flow restrictor installed in each main steamline reactor vessel nozzle:	200% of rated flow After the valve is approximately 75% closed, steam flow is further reduced as a function of the valve area versus travel characteristic
MSIVs longest design closing time:	5.0 seconds
MSIV shortest design closing time:	3.0 seconds
MSIV combined leakage	Combined leakage through the MSIVs for all four MSLs is less than or equal to 66.1 liters per minute at standard temperature of 20°C (68°F) and pressure (one atmosphere absolute pressure) with the differential pressure across the MSIV equal to or greater than 0.269 MpaD (40 psid).
ICS station blackout (i.e., unavailability of all AC power) capability:	≥72 hours
IC sizing:	Sized to remove post-reactor isolation decay heat with three out of four ICs operating and to reduce reactor pressure and temperature to safe shutdown conditions, in 36 hours, with occasional venting of radiolytically generated noncondensable gases to the suppression pool.
ICS Performance Requirements:	Heat removal capacity of the ICS (with 3 of 4 IC trains in service) is at least 101.25 MWt when reactor is above rated operating pressure.
Condensate return valve stroke-open time	≤ 30 seconds with a logic delay time not to exceed 1 second after the opening setpoint is reached.
IC design capacity:	33.75 MWt each IC unit and is made of two identical modules.

Table 5.4-1

Component and Subsystem Design Controls

Component/Subsystem	Control(s)
ICS loop seal:	Assures that condensate valves do not have 285 °C (545 °F) water on one side of the disk and subcooled water [as low as 10 °C (50 °F)] on the other side during normal plant operation, thus affecting leakage during system standby conditions
The design temperature and pressure of the Class 1 portions of the main steam and feedwater lines (same as that of the RPV)	8.62 MPa gauge (1250 psig) and 302°C (576°F)
Number of MSLs	4
Nominal diameter of each MSL	700 mm (28 in)
The main steamline drain subsystem isolation valves are remote-manually operated from the main control room and are closed when reactor power exceeds:	40%.
Number of Feedwater lines and Branch lines to RPV	2 6
Diameter of each Feedwater line and Branch lines to RPV	550 mm (22 in) 300 mm (12 in)
Combined main steam line volume	Combined volume from RPV to the turbine stop valves and steam bypass valves is greater than or equal to 135 m^3.

Table 5.4-2

Isolation Condenser System Component Design Data

Valve	Status Mode (1)	Failure (2)	Actuator type (3)	Size (mm)	Valve type	Location
Steam Inner Isolation	NO	AI	NMO	350	Gate	Steam line
Steam Outer Isolation	NO	AI	MO	350	Gate	Steam line
Condensate Return Inner Isolation	NO	AI	MO	200	Gate	Condensate to RPV
Condensate Return Inner Isolation	NO	AI	NMO	200	Gate	Condensate to RPV
Condensate return	NC	AI	MO	200	Gate	Condensate to RPV
Condensate Return Bypass	NC	FO	NO	200	Globe	Condensate to RPV
Upper Header Vent	NC	FC	SO	25	Globe	Vent line to SP
Upper Header Vent	NC	FC	SO	25	Globe	Vent line to SP
Lower Header Vent	NC	FC	SO	20	Globe	Vent line to SP
Lower Header Vent	NC	FC	SO	20	Globe	Vent line to SP
Lower Header Bypass Vent	NC	AI	MO	20	Globe	Vent line to SP
Lower Header Bypass Vent	NC	AI	MO	20	Globe	Vent line to SP
Purge	NO	AI	MO	20	Globe	Purge line to MSL

Legend:

(1) NO = Normally open; NC = Normally closed;

(2) AI = As is; FO = Fail open; FC = Fail closed;

(3) NMO = Nitrogen rotary motor operated; SO = Solenoid operated;
NO = Nitrogen piston operated; MO = Electric motor operated.

Table 5.4-3

Reactor Water Cleanup/Shutdown Cooling System Data

Number of trains	Two
Demineralizer type	Mixed bed
Demineralizer Capacity (% of rated FW system flow per train)	1
Flow rate per train in Cleanup Mode (one train operation)	116 m^3/hr (512 gpm)
RWCU/SDC shell side RHX exit temperature in Cleanup Mode:	Approximately 226.7°C (440°F).
Maximum allowed cooling water outlet temperature from the NRHX when operated in the shutdown, startup, hot standby, isolation event or overboarding (i.e., dumping water to the main condenser or to the radwaste system) modes:	60°C (140°F).
Flow, through the bottom head connections during heatup and startup operations to prevent thermal stratification (two train operation)	181.6 m^3/hr (800 gpm).
RWCU/SDC flow rate (after heatup) (two train operation)	181.6 m^3/hr (800 gpm)
Approximate flow, during the initial heatup, overboarded to the main condenser (two train operation)	363.2 m^3/hr (1600 gpm) maximum 181.6 m^3/hr (800 gpm) minimum
Approximate maximum flow, during startup overboarded to the main condenser	112.2 m^3/hr (494.2 gpm)
The combined system process flow range from the bottom drain line and the RPV mid-region nozzle suction line (per train)	90.8 m^3/hr (400 gpm) to 682.6 m^3/hr (3005.5 gpm).
RWCU/SDC shutdown cooling design maximum flow rate (two train operation)	1365.2 m^3/hr (6011 gpm).

Table 5.4-3

Reactor Water Cleanup/Shutdown Cooling System Data

RWCU/SDC system shutdown cooling function heat removal capacity (two train operation)	55.4 MWt (189.2 MBtu/hr).
From the rated design pressure and temperature, in conjunction with the heat removal capacity of either the main condenser and/or the isolation condensers, the time to cool down the reactor coolant temperature to - 60°C (140°F) - 54°C (130°F) - 49°C (120°F)	 24 hours 40 hours 96 hours

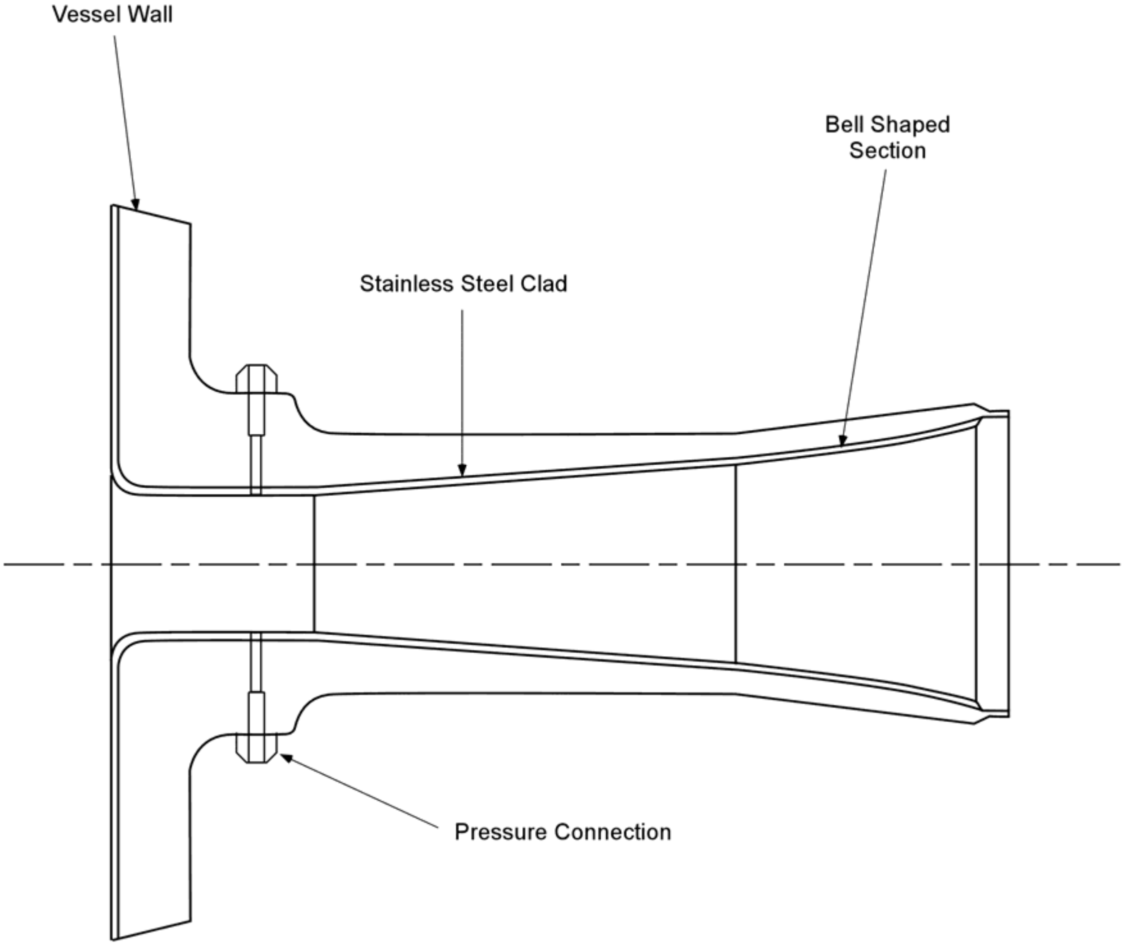


Figure 5.4-1. Main Steamline Nozzle and Flow Restrictor

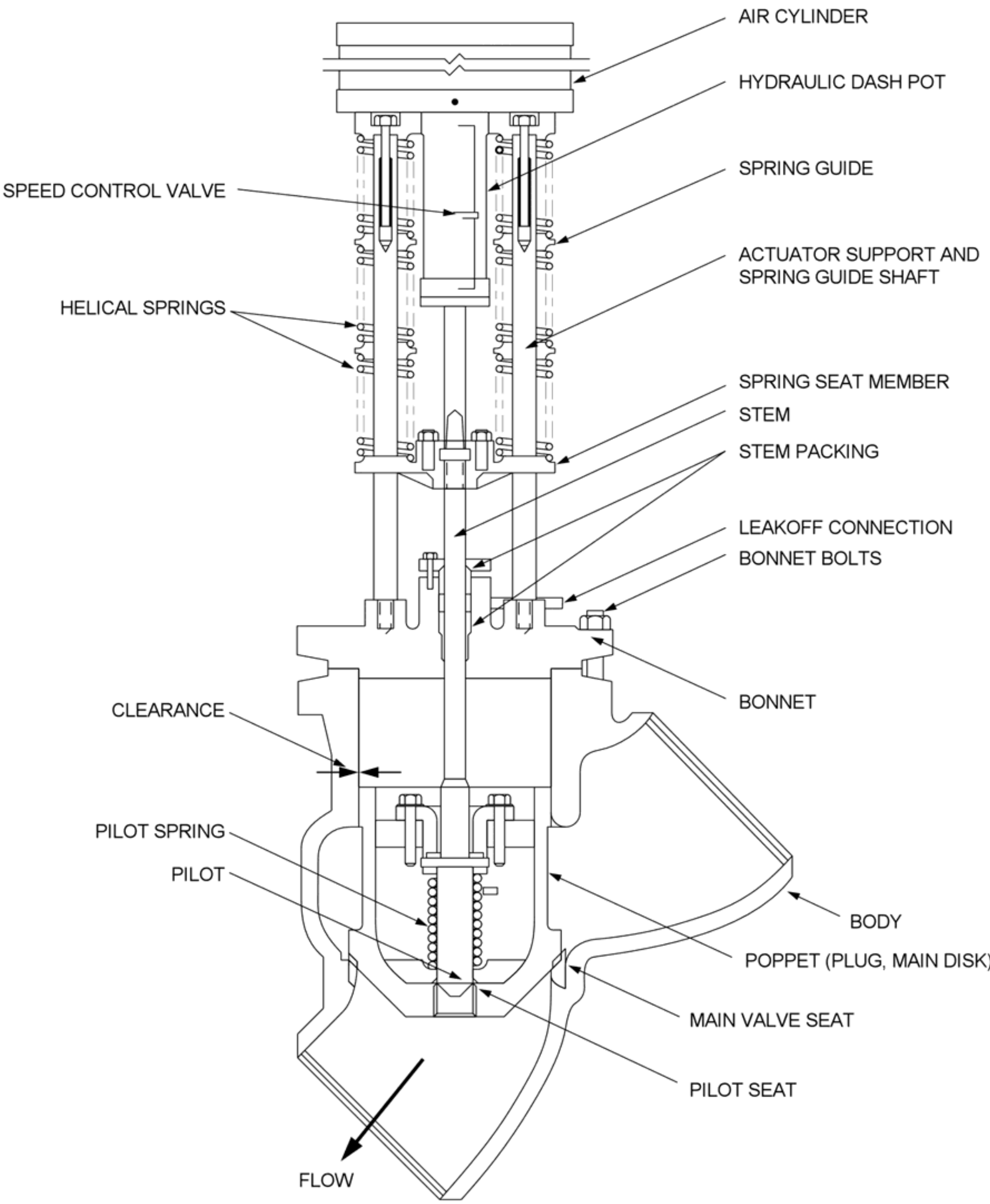


Figure 5.4-2. Main Steamline Isolation Valve

Figure 5.4-3. Layout of Main Steam and Feedwater Lines

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}